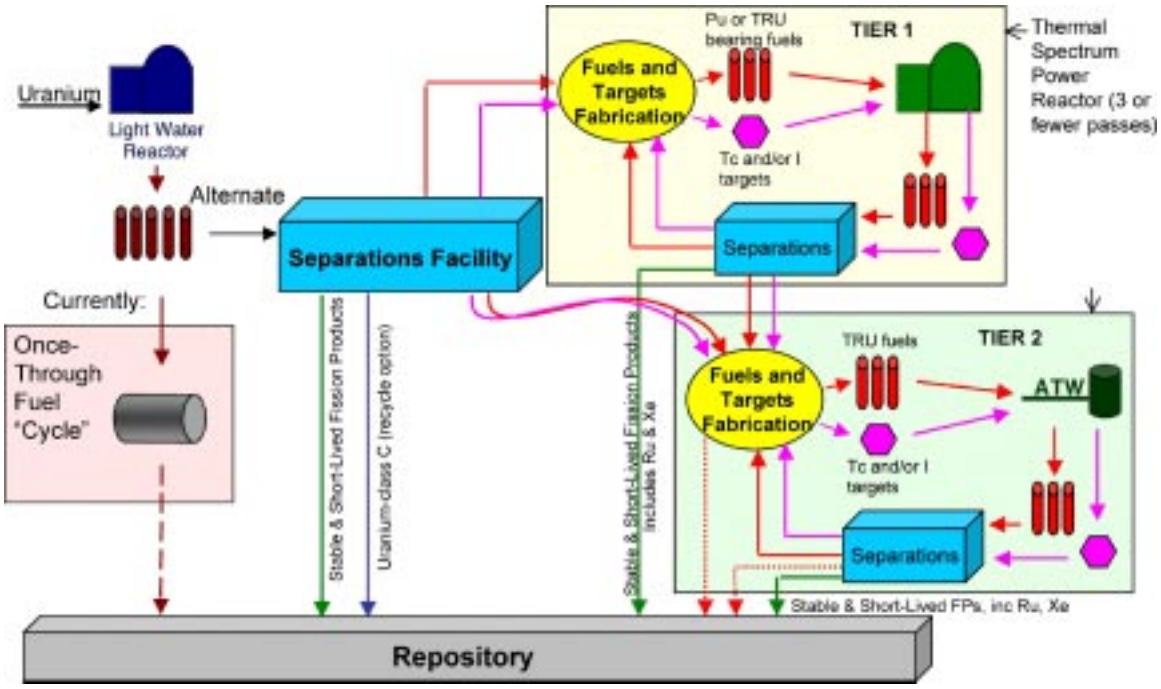
PDO-RGN-1009
AAA-PDO-GEN-01-0051
LA-UR-01-5572
September 2001
Revised November 2001

Candidate Approaches for an Integrated Nuclear Waste Management Strategy—Scoping Evaluations



Pre-Decisional Information

Candidate Approaches for an Integrated Nuclear Waste Management Strategy—Scoping Evaluations

PDO-RGN-1009
AAA-PDO-GEN-01-0051
LA-UR-01-5572
September 2001
Revised November 2001

Author

Signature on file

AAA-PDO — Gregory J. Van Tuyle, November 2001

Recommended for Approval:

Project Director — *Signature on file*

AAA-PDO — R. Bruce Matthews, November 2001

Report Authors

Gregory J. Van Tuyle (Los Alamos National Laboratory), Phillip Finck (Argonne National Laboratory)

Contributors

Alan Baxter (General Atomics), Deborah Bennett (LANL), Douglas Crawford (ANL-West), Ken Chidester (LANL), J. Wiley Davidson (LANL), Russ Durrer (LANL), Diane Graziano (ANL), William G. Halsey (Lawrence Livermore National Laboratory), Robert Hill (ANL), James Laidler (ANL), Denise Pelowitz (LANL), Massimo Salvatores (CEA, France), Shelly Spearing (LANL), John Stillman (ANL), Temitope Taiwo (ANL), Michael Todosow (Brookhaven National Laboratory), Holly Trellue (LANL), Mark Williamson (ANL)

Technical Editors

Berylene Rogers and Scott Milder (LANL)

Acknowledgments

The report reflects the efforts of a dedicated team within the Advanced Accelerator Applications Program, many of whom also contributed to the drafting of this report. The effort was carried forward under the guidance of a steering committee that included Mark Williams (Department of Energy), Bruce Matthews and Ed Arthur (LANL), and Dave Hill (ANL). Greg Van Tuyle (LANL) and Phillip Finck (ANL) managed the effort. Robert Hill (ANL) and Deborah Bennett (LANL) provided technical leadership. Many people contributed to the technical evaluation effort, including the following: Alan Baxter (GA), Douglas Crawford (ANL-W), Ken Chidester (LANL), J. Wiley Davidson (LANL), Mike Fikani (GA), Diane Graziano (ANL), William G. Halsey (LLNL), James Laidler (ANL), Denise Pelowitz (LANL), Shelly Spearing (LANL), John Stillman (ANL), Temitope Taiwo (ANL), Michael Todosow (BNL), Holly Trellue (LANL), Mark Williamson (ANL), Chuck Bathke (LANL), and Bob Krakowski (LANL). Within a five-month time period this team completed an effort that could have normally been expected to continue for a year or longer.

Several others contributed to this report, either through written feedbacks or via comments during key discussions. Ron Omberg (Pacific Northwest National Laboratory) actively participated in the editing effort on DOE's behalf to ensure the report addressed most of the pertinent questions. Many of those questions (and recommendations) were posed by Dr. Burton Richter (Stanford Linear Accelerator Center) and the NERAC Subcommittee [Richter, Omberg, Darleane Hoffman (Lawrence Berkeley National Laboratory. University of California, Berkeley), Joy Rempe (Idaho National Engineering and Environmental Laboratory), Sekazi Mtingwa (North Carolina Agriculture and Technology College), Susan Ion (British Nuclear Fuels Limited), and Victor Gilinsky (retired NRC)] that advises DOE on this program. Of the DOE staff, in addition to Mark Williams, John Herczeg and Frank Goldner contributed helpful suggestions throughout this team's efforts. Finally, Takehiko Mukaiyama (JAERI/Japan), Massimo Salvatores (CEA/France), and John Lee (University of Michigan) all contributed to some very useful discussions during a July workshop on the potential of fast-spectrum vs. thermal-spectrum systems for waste transmutation. All these contributions from individuals with valuable insights are deeply appreciated.

Distribution List

AAA Project File	H816
AAA-RMDC	C341
Angelos, Jimmy	SRS
Arthur, Ed	F606
Bathke, Chuck	F607
Baxter, Alan	C341
Bennett, Deborah	H816
Boore, Brent	SRS
Bresee, Jim	DOE/NE-20
Cappiello, Mike	H816
Carrelli, Mario	Westinghouse, Pittsburgh
Chidester, Ken	H816
Crawford, Douglas	ANL-W
Davidson, Wiley	F607
Durrer, Russ	K551
Espinosa, Sam	DOE/AL
Fikani, Mike	C341
Finck, Phillip	ANL
Gilinsky, Victor	Retired, NRC
Gohar, Yousry	ANL
Goldner, Frank	DOE/NE-20
Graziano, Diane	ANL
Greenspan, Ehud	UC-Berkeley
Haberman, Norton	DOE/NE-20
Halsey, William	LLNL
Hartline, Bev	ANL
Herczeg, John	DOE/NE-20
Hermannsfeld, William	SLAC
Hill, Dave	ANL
Hill, Robert	ANL
Hoffman, Darleane	LBNL/UCB
Ion, Susan	BNFL
Khalil, Hussein	ANL
Krakowski, Bob	F607
Laidler, James	ANL
Lee, John C.	U of Mich
Matthews, Bruce	H816
McConnell, Steve	H816
Metzler, Jack	DOE/DP
Mtingwa, Sekazi	NC A&T
Mukaiyama, Takehiko	JAERI/Japan
Omberg, Ron	PNNL
Pasamehmetoglu, Kemal	H816
Pelowitz, Denise	K575
Quintana, Lawrence	H816
Rempe, Joy	INEEL
Richter, Burton	SLAC
Rodriguez, Carmello	C341
Roglans. Jordi	ANL
Salvatores, Massimo	CEA, France
Schneider, David	H816
Spearing, Shelly	H836
Stillman, John	ANL
Stoll, Wolfgang	Consultant
Taiwo, Temitope	ANL
Thomas, Kim	J515
Todosow, Michael	BNL
Torres, JoAnn	H816
Trellue, Holly	K575
Van Tuyle, Greg	H816
Ward, Tom	DOE/NE-20
Waters, Laurie	K575
Williams, Mark	DOE
Williamson, Mark	ANL

Table of Contents

List of Figures

List of Tables

List of Acronyms

AAA	Advanced Accelerator Applications
ADS	Accelerator-Driven System
ALMR	Advanced Liquid Metal Reactor
Am-242m	Metastable isotope of americium
ANL	Argonne National Laboratory
ATW	Accelerator-driven Transmutation of Waste
BNFL	British Nuclear Fuels Limited
BNL	Brookhaven National Laboratory
BOEC	Beginning-of-equilibrium-cycle
BOL	Beginning-of-life
BWR	Boiling water reactor
CANDU	Canada Deuterium Uranium
CEA	Commissariat à l'énergie atomique, the French Atomic Energy Commission
CSNF	Commercial spent nuclear fuel
CWF	Ceramic waste form
D	Diameter
DIF3D	Neutronics code for finite-difference diffusion theory problem solving
DOE	US Department of Energy
DRAGON	A lattice code
DSNF	DOE-owned spent nuclear fuel
EBR-II	Experimental Breeder Reactor II
efpd	Effective full-power days
EOC	End-of-cycle
EOEC	End-of-equilibrium-cycle
ES&H	Environment, safety, and health
eV	Electron volt
FFTF	Fast-Flux Test Facility
FP	Fission product
FR	Fast reactor
GA	General Atomics
GCFR	Gas-Cooled Fast Reactor
GeV	Giga-electron volt

GT/AD-MHR	Gas Turbine/Accelerator-Driven Modular Helium Reactor
GT-MHR	Gas Turbine Modular Helium Reactor
GWt	Gigawatt-thermal—a unit of thermal power
HFR	High flux reactor
HLW	High-level radioactive waste
HM	Heavy metal
IFR	Integral Fast Reactor
INEEL	Also INEL; Idaho National Engineering and Environmental Laboratory
JAERI	Japan Atomic Energy Research Institute
JEF	Joint Evaluated File—library of neutron reaction data
kgHM	Kilograms of heavy metal
LANL	Los Alamos National Laboratory
LBNL	Lawrence Berkeley National Laboratory
LBE	Lead-bismuth eutectic
LLFP	Long-lived fission product
LLNL	Lawrence Livermore National Laboratory
LWR	Light-water reactor
MA	Minor actinides
MC^2-2	A code to calculate fast neutron spectra and multi-group cross sections
MCNP	A Monte Carlo N-Particle code
MeV	Million electron volts
MHTGR	Modular High-Temperature Gas Reactor
MOX	Mixed-oxide
MT	Metric ton
MTHM	Metric ton of heavy metal
MW	Megawatt or million watts—a unit of power
MWd	Megawatt days
MWF	Metal waste form
MWt	Megawatt thermal—a unit of thermal power
n/cm^2	Neutrons per square centimeter—a unit of neutron fluence
NCAT	North Carolina Agriculture and Technology College
NEA	Nuclear Energy Agency
NEPD	National Energy Policy Development Group
NERAC	Nuclear Energy Research Advisory Committee

NFF	Nonfertile fuel
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
ORIGEN	A computer code system for calculating the buildup, decay, and processing of radioactive materials
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PRISM	Power Reactor Innovative Small Module Advanced Liquid Metal Reactor
PSI	Paul Scherrer Institute
PUREX	Plutonium Uranium Extraction—commonly used process based on using tributylphosphate for fuel reprocessing
PWR	Pressurized water reactor
R&D	Research and Development
REBUS-3	Reactor fuel-cycle analysis code
RG	Reactor grade
RMDC	Records Management Document Control
RPO	Research Project Office
SLAC	Stanford Linear Accelerator Center
SNF	Spent nuclear fuel
SPD	System Point Design
STATS	Separations Technology and Transmutation Systems (a committee of the National Research Council)
Sv/Bq	Sieverts per becquerel
TRISO	Tri-isotropic, referring to a multi-layered fuel-particle coating consisting of pyrolytic carbon and silicon carbide
TRU	Transuranic element
TSPA-SR	Total System Performance Assessment—Site Recommendation
UOX	Uranium oxide
UREX	Uranium extraction—an aqueous process for processing spent fuel
VARIANT	A nodal transport and diffusion module for DIF3D
WIMS8	Winfrith Improved Multigroup Scheme, a general code for reactor lattice cell calculation on a wide range of reactor systems
YMP	Yucca Mountain Project

Executive Summary

To support the long-term growth of nuclear power as a significant portion of US generating capacity, the Department of Energy is exploring options for an integrated nuclear waste management strategy. This strategy uses partitioning and transmutation processing of spent fuel to create transmutation fuels and targets to be burned in power reactors and/or accelerator-driven systems, thereby eliminating the most problematic components of nuclear waste (also see Appendix A). This concept is known as a multi-tier or multi-strata system, in which the first tier includes future power-generating reactors and the second tier includes fast-spectrum systems. The multi-tier approach is being evaluated and contrasted against traditional, single-tier transmutation systems based on fast-spectrum reactors or accelerator-driven subcritical systems. The US government has responsibility for the disposal of spent nuclear fuel; therefore, the waste disposal mission of the transmutation system would be government sponsored, although it could use commercial and government facilities.

The multi-tier transmutation system is based on the premise that a large portion of the nuclear waste stream can be transmuted in commercial power reactors while producing energy to offset the cost of transmutation. This first-tier transmutation would also reduce the size and probably the costs of dedicated second-tier transmutation systems. The European and Japanese programs in nuclear waste partitioning and transmutation consistently relegate much of the plutonium to nuclear power reactors, focusing the dedicated transmuters upon minor actinides, i.e., the heavy isotopes of plutonium and higher elements, and some fission products (see Appendix B for discussion of international studies related to multi-tier approach). Routine recycle of plutonium into commercial nuclear reactors has been precluded in the US by presidential directive since the mid-1970s. However, the option of plutonium consumption in commercial reactors was re-introduced during the Clinton Administration as an option for destroying excess plutonium from the weapons program.

Although policy issues remain, the possibility of destroying plutonium and other hazardous radioactive isotopes in commercial spent fuel using nuclear power reactors presents a viable technical option that has been demonstrated commercially. The multi-tier approaches evaluated in this study include systems in which plutonium (with uranium and/or minor actinides in some approaches) first passes through first-tier thermal-spectrum power reactors, with the residuals being subsequently passed to second-tier fast-spectrum transmuters. Single-tier systems, in which all transuranics are fissioned in a single machine, are also evaluated.

The choice of technologies affects transmutation performance. The transmutation performance of light-water cooled reactors, which are widespread today and for the foreseeable future, is different from the performance of gas-cooled reactors, which could be widespread in the future. Similarly, reactor fuels that contain minor actinides perform differently than those that do not. And finally, different chemical and physical types of reactor fuels, such as oxide fuel as compared to metal fuel, perform differently. Furthermore, the technology and fuel used in the first-tier reactor system affects the efficiency of the transmutation in the second tier. Because of these technology choices, this report evaluates seven different two-tier reactor and fuel technologies, along with a single-tier fast-reactor system and a single-tier accelerator-driven system.

In considering the results of the current scoping study, and the cross-comparison of the performance of the candidate approaches, it must be understood that the search is not for the best approach. Rather, it is a search for viable approaches and an understanding of the implications of such approaches on the technology development needs.

The Advanced Accelerator Applications (AAA) Program has recently established an initial set of high-level programmatic goals for the nuclear waste transmutation mission. Each top-level goal is

supported by specific programmatic criteria. Taken in their entirety, this initialset of goals and criteria addresses most, if not all, of the concerns expressed by the National Academy of Science STATS (Separations Technology and Transmutation Systems) panel review of transmutation options (1996), as well as those concerns attributed to independent groups opposed to this technology option. Although some of the criteria support more than one high-level goal, they are grouped as follows:

Improve public safety

- Radiotoxicity Criterion: Reduce radiotoxicity of spent nuclear fuel below that of source uranium within a few thousand years. [Note: For this study, 1,000 years was used.]
- Dose Criterion: Reduce maximum predicted peak dose to future inhabitants of a region containing a repository by at least 99% in comparison to current predictions.

Provide benefits to the repository program

- Heat-Load Criterion: Reduce long-term heat load of spent nuclear fuel by at least 90% after 500 years as compared to unprocessed spent fuel.
- Criticality Criterion: Preclude possibility of future criticalities by reducing and degrading the transuranic content.
- Mass Criterion: Reduce mass of commercial spent fuel by separating the uranium and either recycling the uranium or diverting it to alternate disposal.

Reduce the proliferation risk from plutonium in commercial spent fuel

- Plutonium Inventory Criterion: Reduce or potentially reverse the buildup of the inventory of plutonium in nuclear fuel cycle, reversing the long-term trend of plutonium build-up from the once-through fuel cycle.
- Plutonium Disposal Criterion: Reduce the inventory of plutonium passing to the nuclear waste repository by 99% and decrease the fissile fraction within that plutonium.
- Plutonium Accessibility Criterion: Minimize the risk of plutonium diversion throughout the alternate fuel-cycle and materials-handling processes.

Improve prospects for nuclear power

- Viability Criterion: Provide a viable and economically feasible waste management option for commercial spent nuclear fuel.
- Technical Risk Criterion: Minimize technical risk to achieve solutions to nuclear waste challenge.
- ES&H Criterion: Improve upon ES&H characteristics of the once-through fuel cycle.

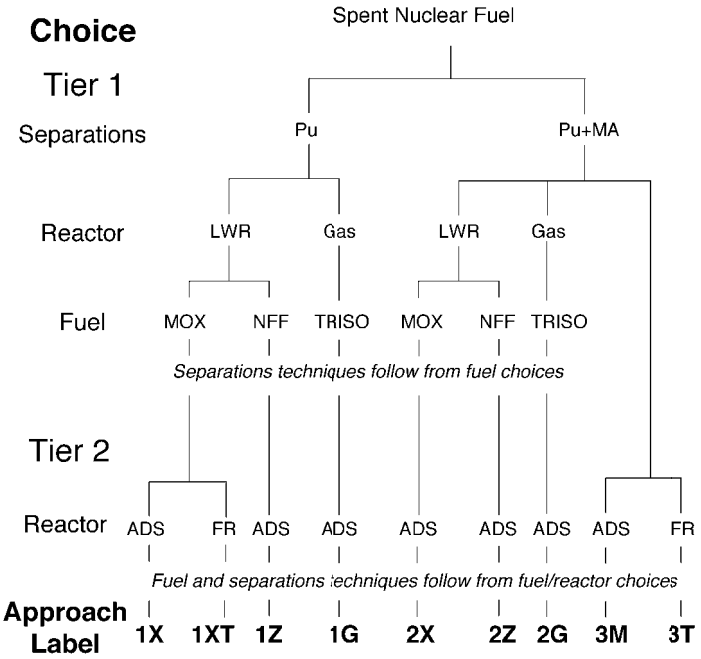
Several waste management scenario studies have been analyzed in the past 20 years; these have generally been lengthy efforts run in large international contexts. The criteria of this study are quite different: it was our purpose to obtain in a short time a top-level understanding of the major consequences of technology choices with respect to the ability of the various approaches to meet the criteria of the AAA Program.

Thus, the objective of this study consists of generating isotopic mass flows and waste stream flows for each scenario; these mass flows are then evaluated to estimate doses to the public and workers, radiotoxicities, fissile flows, waste volumes, and heat loads. These quantities provide the basis for comparing the performances of all approaches with respect to the AAA Program criteria.

For the scoping evaluations we assumed three primary approaches for considering multi-tier alternatives to spent nuclear fuel (SNF) transmutation: Approach 1 assumed plutonium-fueled thermal

spectrum reactors in Tier 1, with minor actinides and Tier 1 residuals going directly to Tier 2 for fast-spectrum systems to complete the transmutation; Approach 2 assumed transuranic (TRU) elements-fueled Tier 1 thermal spectrum reactors with residuals going to the Tier 2 fast-spectrum systems; Approach 3 assumed all SNF transuranics are transmuted only in fast-spectrum systems with different conversion ratios. Approach 1 was derived from the dual-strata approach considered the reference case in Japan and France. Nevertheless, whereas these foreign programs consider a combination of thermal and fast systems to burn the plutonium and an accelerator-driven system to burn the minor actinides, the AAA Program considers that the development of three distinct technologies would be too burdensome and decided to rely only on thermal reactors to burn plutonium as far as possible. Approach 2 is similar to Approach 1, but transuranics are not separated. Approach 3 consists of following either the Accelerator-driven Transmutation of Waste (ATW) approach, in which transuranics are transmuted in an accelerator-driven system without any fertile material, or the fast reactor approach, in which transuranics are mixed with fertile material, thus allowing the safe operation of a critical reactor.

Approach Tree



Labeling scheme for the nine approaches considered.

At this point in the evaluation process, the following principal conclusions can be drawn:

- All of the assessed approaches can fundamentally meet the transmutation criteria as stated, within the fidelity of the available data, assumptions, and analytical methods. It appears technically feasible to move toward an integrated waste management strategy using future reactors.

- Virtually every criterion, and therefore the transmutation performance of each approach, is most significantly influenced by the ability to achieve 99.9% separation of TRU materials from spent nuclear fuel.
- The current state of knowledge regarding separations and fuel fabrication losses is quite limited: while industrial scale plutonium uranium extraction (PUREX) plants have systematically achieved separation losses equal to or lower than 0.1% for uranium-oxide fuel, and to a very limited extend for mixed-oxide fuels, other processes, particularly those designed for as yet undeveloped fuel types, cannot yet be assessed with precision. Thus, detailed sensitivity studies must accompany the development of flow sheets, and a major research and development effort is required to develop and demonstrate the separations and fabrication technologies.
- Another important factor is the achievable burnup rates, which also strongly affect the overall transmutation losses (by changing the number of separations passes) and require an extensive fuel development and demonstration program.
- For two-tier systems, there is a clear advantage to trying to maximize the overall burnup rate in the first tier. As mentioned, theoretical studies indicate high potential burnup rates, but practical considerations usually limit the achievable rate. The issue needs to be studied carefully, taking into account all practical considerations in the fuel cycle.

Regarding the performance of candidate approaches vs. the goals, criteria, and metrics, we can conclude:

- Each assessed approach can reduce the radiotoxicity of spent nuclear fuel to below the radiotoxicity of natural uranium ore within 1,000 years, assuming 0.1% separation losses.
- From a qualitative approach, it appears that each assessed approach can reduce maximum predicted peak dose to future inhabitants by at least 99% in comparison to current predictions.
- Each assessed approach can reduce the inventory of materials that contribute to long-term heat loads in the repository by 90% or more.
- Each assessed approach reduces the transuranic mass by greater than 99%, under the given assumptions, but each cases exhibits a percentage increase in fission product mass. Likewise, there is a sharp percentage increase in TRU and fission product waste volume, but this condition can be alleviated by re-assessing the assumption for two year cooling rather than a longer period. It should be noted that appropriate disposition paths for graphite fuel element material must be explored.
- Each assessed approach reduces plutonium inventory by greater than 99%, which certainly exceeds the 90% nominal basis.
- The support ratio assessments clearly demonstrate that scope of the enterprise necessary to support a nuclear future assumption, and are variable dependent on assumptions regarding electricity production at each tier.

This evaluation has provided a set of preliminary conclusions that are strictly limited by the overriding assumptions. However, the robustness of performance in light of these assumptions and their variability, knowledge enhancement through R&D, and more comprehensive, and systematic analytical approaches must be verified in future analyses as noted here:

- Assess economic performance of approaches with attention to support ratios.

- Assess the sensitivity of transmutation performance to variations in assumptions, especially the 99.9% separation efficiency assumption.
- Perform R&D on process factors that most significantly impact separation efficiency, and seek process approaches that ensure efficient separations.
- Perform R&D on process factors that most significantly impact the efficiency of fuel processing, and seek process approaches that ensure efficient fuel fabrication.
- Determine the maximum achievable Tier 1 burnup in balance with radiotoxicity and dose of materials sent to the repository, as well as increases in fission product inventories and waste volumes.
- Assess uncertainty incurred by lack of data, assumptions, and analytical methods.
- Assess alternate approaches, and variations on current approaches.

Candidate Approaches for an Integrated Nuclear Waste Management Strategy—Scoping Evaluations

1. Introduction

To support the long-term growth of nuclear power as a significant portion of US generating capacity[1], the Department of Energy is exploring options for an integrated nuclear waste management strategy. This strategy uses partitioning and transmutation processing of spent fuel to create transmutation fuels and targets to be burned in power reactors and/or accelerator-driven systems, thereby eliminating the most problematic components of nuclear waste (see Appendix A). This concept is known as a multi-tier or multi-strata system, in which the first tier includes future power-generating reactors and the second tier includes accelerator-driven systems. The multi-tier approach is being evaluated and contrasted against traditional, single-tier transmutation systems based on fast-spectrum reactors or accelerator-driven subcritical systems. The US government has responsibility for the disposal of spent nuclear fuel; therefore, the waste disposal mission of the transmutation system would be government sponsored, although it could use commercial and government facilities.

The current US effort to develop nuclear waste partitioning and transmutation technologies was launched via the ATW (Accelerator-based Transmutation of Waste) Road Map [1] effort during fiscal year 1999. Although there were earlier efforts [2,3,4] within the US to propose such a technological approach, as well as efforts to evaluate the potential of such systems [5], the US effort lagged far behind efforts in France [6] and Japan [7], for example. Thus, the 1999 ATW Road Map effort represented both an opportunity and a challenge, in that the US situation regarding spent nuclear fuel, nuclear fuel cycles, and possible technology implementation scenarios had not been analyzed systematically. Rather, various technology options[2] were studied and advocated as stand-alone solutions to the well-known nuclear waste dilemma. The 1999 ATW Road Map focused primarily on one technology option: accelerator-driven fast neutron spectrum transmutation systems and one implementation scenario: burn-down of the spent fuel from all past and existing US power reactors. Consequently, the proposed system is large and costly, the mission time is long, and the technical and economic challenges are formidable. A revision of the 1999 ATW Road Map is to begin in the fall of 2001.

The multi-tier transmutation system is based on the premise that a large portion of the nuclear waste stream can be transmuted in commercial power reactors while producing energy to offset the cost of transmutation. This first-tier transmutation would also reduce the size and probably the costs of dedicated second-tier transmutation systems. The European and Japanese programs in nuclear waste partitioning and transmutation consistently relegate much of the plutonium to nuclear power reactors, focusing the dedicated transmuters upon minor actinides, i.e., the heavy isotopes of plutonium and higher elements,

[1]Report of the National Energy Policy Development Group, May 2001: “The NEPD Group recommends that the President support the expansion of nuclear energy in the United States as a major component of our national energy policy.”

[2]The US nuclear power generating industry is a commercial enterprise, unlike French industry, and the US government has the ultimate responsibility for the nuclear waste. The companies that comprise the US commercial enterprise can take different directions on technology for business reasons. Consequently, if the government desires to promote integrated waste management strategies, it needs to develop a variety of technologies and alternative approaches. These must be capable of dealing with various reactor technologies and fuel characteristics that the industry chooses, as well as the social, political, economic and other environmental factors that would bear on processing and transportation systems.

and some fission products (see Appendix B for discussion of international studies related to multi-tier approach). Routine recycle of plutonium into commercial nuclear reactors has been precluded in the US by presidential directive since the mid-1970s. However, the option of plutonium consumption in commercial reactors was re-introduced during the Clinton Administration as an option for destroying excess plutonium from the weapons program.

Although policy issues remain, the possibility of destroying plutonium and other hazardous radioactive isotopes in commercial spent fuel using nuclear power reactors presents a viable technical option that has been demonstrated commercially. Therefore, the multi-tier approaches evaluated in this study comprise systems in which plutonium (perhaps with uranium or minor actinides) would first pass through first-tier thermal-spectrum power reactors, and the residuals would subsequently pass into the dedicated, fast-spectrum, second-tier transmuters. The study also evaluates single-tier systems in which all transuranics are fissioned in a single machine. The choice of technologies affects transmutation performance. The technology of the first-tier reactor system and the transmuter fuel affects the efficiency of the transmutation in the first tier. The transmutation performance of light-water cooled reactors, which are widespread today and for the foreseeable future, is different from the performance of gas-cooled reactors, which could be widespread in the future. Similarly, transmuter fuels that contain minor actinides perform differently than those that do not. And finally, different chemical and physical types of transmuter fuels, such as oxide fuel when compared to metal fuel, perform differently. Because of these technology choices, this report evaluates seven different two-tier reactor and fuel technologies, along with a single-tier fast-reactor system and a single-tier accelerator-driven system. The performance of these systems is then compared to top-level program goals and criteria to compare overall transmutation mission performance.

Top-level programmatic goals include the following:

- Improve public safety;
- Provide benefits to the repository program;
- Reduce the proliferation risk from plutonium in commercial reactor spent fuel; and
- Improve prospects for nuclear power.

Each top-level goal is supported by program criteria that are quantifiable, as discussed in Chapter 2 of this report. By evaluating the predicted performance of proposed multi-tier approaches vs. the quantifiable criteria it is possible to assess the viability of candidate approaches. It is more difficult and perhaps unwise to rank the performance of the myriad of candidate technologies, given the fact that market forces will actually drive developments within the nuclear power sectors. Of greater value is an assessment as to the likely performance of the approaches deemed more probable in order to scope out the anticipated performance of such systems.

There are many possible variations of thermal-spectrum reactors, fast-spectrum systems, fuel forms and fabrication techniques, separations techniques, and waste forms, so a complete analysis of all variations would require a large and extended effort. Instead, approaches of particular interest are currently being evaluated, and some variations on those approaches are to be worked next year. Three major approaches are under evaluation, to be compared against each other and the once-though cycle. The first approach assumes separated plutonium is fissioned, to the extent possible, in light-water or gas-cooled thermal-spectrum reactors, using either mixed-oxide or nonfertile fuels. The second approach assumes transuranics (no plutonium separation) are passed through gas-cooled or light-water reactors. The third approach entails driving the entire transuranic and fission product waste stream through fast-spectrum transmutation systems, either subcritical (as in the 1999 ATW Road Map) or critical.

Predictions of mass flows (including isotopics) through the multi-tier systems for approaches of interest are presented in Chapter 5. Based on those predictions, the toxicity, dose, proliferation, mass, heat load,

and other parameters of importance were analyzed. This facilitated evaluation of the various approaches utilizing the system performance criteria, and which can also indicate the relative performances of the postulated approaches, as discussed in Chapter 6. The next phase of the planned effort to evaluate multi-tier systems is described in Chapter 7, and the report is summarized and conclusions are drawn in Chapter 8.

In the nuclear power system architecture employed in this study, Tier 1 systems are considered to function primarily as power producers, and their deployment will be driven in part by market forces. The Tier 2 systems must be dedicated transmutation systems, although the fission energy could be harnessed and used to defray costs. Thus, in developing multi-tier transmutation systems, much of the focus will be on understanding the implications of potential Tier 1 systems and developing the Tier 2 system to be responsive to the various possibilities for Tier 1. If a Tier 1 system is particularly attractive for waste transmutation, this could become an important factor in the system being utilized, but it should not be the only consideration. Therefore, in considering the results of the current scoping study, and the cross-comparison of the performance of the candidate approaches, it must be understood that the search is not for the best approach. Rather, it is a search for viable approaches and an understanding of the implications of such approaches on the technology development needs.

2. Program Goals, Criteria, and Metrics

The Advanced Accelerator Applications (AAA) Program has recently established a set of high-level programmatic goals for the nuclear waste transmutation mission. These high-level goals were then used to generate specific programmatic criteria to judge the effectiveness and attractiveness of candidate transmutation systems, and also to establish performance goals for technologies important to the transmutation mission.

2.1. Top-Level Programmatic Goals

Through its interactions with the Nuclear Energy Reactor Advisory Committee (NERAC) Subcommittee on Accelerator Transmutation of Waste (recently renamed the Subcommittee on Advanced Nuclear Transformation Technology) [1], the AAA Program, which includes the ATW effort, has established the following set of four top-level goals:

- Improve public safety,
- Provide benefits to the repository program,
- Reduce the proliferation risk from plutonium in commercial spent fuel, and
- Improve prospects for nuclear power.

For each high-level goal, there is potential for contributing to these goals through implementation of an effective and efficient nuclear waste transmutation system. However, there are also instances where such an approach could have adverse consequences. In these approaches, there is the need to specify criteria to avoid the potentially adverse impacts of an ill-conceived approach. For example, in addressing the proliferation risk criterion, it is important to minimize any near-term vulnerability to nuclear materials diversions while reducing the long-term vulnerabilities associated with managing large inventories of plutonium.

2.2. Programmatic Criteria

The AAA Program has recently established a set of high-level programmatic goals for the nuclear waste transmutation mission. Each of these top-level goals is supported by specific programmatic criteria. Taken in their entirety, this set of goals, criteria, and metrics addresses most, if not all, of the concerns expressed by the National Academy of Science STATS panel review of transmutation options (1996), as well as those concerns attributed to independent groups opposed to this technology option. Although some of the criteria support more than one high-level goal, they are grouped as follows:

I. Improve public safety

I.1. Radiotoxicity Criterion: Reduce radiotoxicity of spent nuclear fuel below that of source uranium within a few thousand years. [Note: For this study, 1,000 years was used.]

I.2. Dose Criterion: Reduce maximum predicted peak dose to future inhabitants of a region containing a repository by at least 99% in comparison to current predictions.

II. Provide benefits to the repository program

II.1. Heat-Load Criterion: Reduce long-term heat load of spent nuclear fuel by at least 90% after 500 years as compared to unprocessed spent fuel.

II.2. Criticality Criterion: Preclude possibility of future criticalities by reducing and degrading the transuranic content.

II.3. Mass Criterion: Reduce mass of commercial spent fuel by separating the uranium and either recycling it or diverting it to alternate disposal.

III. Reduce the proliferation risk from plutonium in commercial spent fuel

III.1. Plutonium Inventory Criterion: Reduce or potentially reverse the buildup of the inventory of plutonium in nuclear fuel cycle, reversing the long-term trend of plutonium build-up from the once-through fuel cycle.

III.2. Plutonium Disposal Criterion: Reduce the inventory of plutonium passing to the nuclear waste repository by 99% and decrease the fissile fraction within that plutonium.

III.3. Plutonium Accessibility Criterion: Minimize the risk of plutonium diversion throughout the alternate fuel-cycle and materials-handling processes.

Improve prospects for nuclear power

IV.1. Viability Criterion: Provide a viable and economically feasible waste management option for commercial spent nuclear fuel.

IV.2. Technical Risk Criterion: Minimize technical risk to achieve solutions to nuclear waste challenge.

IV.3. ES&H Criterion: Improve upon ES&H characteristics of the once-through fuel cycle.

The goal of improved public safety is focused primarily on the repository and its contents, although the ES&H criterion also supports this top-level goal. The inspiration for the nuclear waste partitioning and transmutation approach traces largely to charts that show radiotoxicity of spent nuclear fuel falling to uranium ore levels within 300 years when *all* of the actinides are removed. Given man's ability to create containers and barriers that can survive for thousands of years, this immediately suggests the possibility of easing or simplifying the requirements for the required long-term waste isolation. By removing and transmuting over 99.5% of the transuranics, the radiotoxicity of the remainder will reach the source uranium ore toxicity within a few thousand years. The idealized correlation between the fraction of transuranics removed from the waste stream and radiotoxicity is illustrated in Figure 2-1. These ideal curves present the concept that removal of most of the actinides would cause the radiotoxicity of the commercial spent nuclear fuel to fall below that of uranium ore within a few thousand years. In addition to radiotoxicity considerations, there are a few waste stream components that tend to leach out and be transported into the environment. For the proposed Yucca Mountain repository, technetium, iodine, and neptunium are dominant dose contributors (see Figure 2-2). The Dose Criterion targets these materials for transmutation, with any nontransmuted materials to be placed in long-lived, leach-resistant waste forms.

The high-level goal of benefiting the repository program is supported by three programmatic criteria. The heat-load criterion traces to a current challenge for repository designers, i.e., minimizing any impact of decay heat in spent fuels raising repository temperatures into ranges where unforeseen changes are possible. The criticality criterion traces to the remote possibility that water and favorable geometries could conspire to set up a future criticality in the repository (although improbable, such a criticality is considered far more likely than damaging supercriticalities that were postulated a few years ago [2]). Because possible criticalities force loading constraints, elimination of such constraints should be helpful. But the biggest driver toward a simpler, cheaper repository could be the mass criterion, which involves doing something else (e.g., recycling or disposal elsewhere) with the uranium content, which makes up more than 90% of the spent fuel.

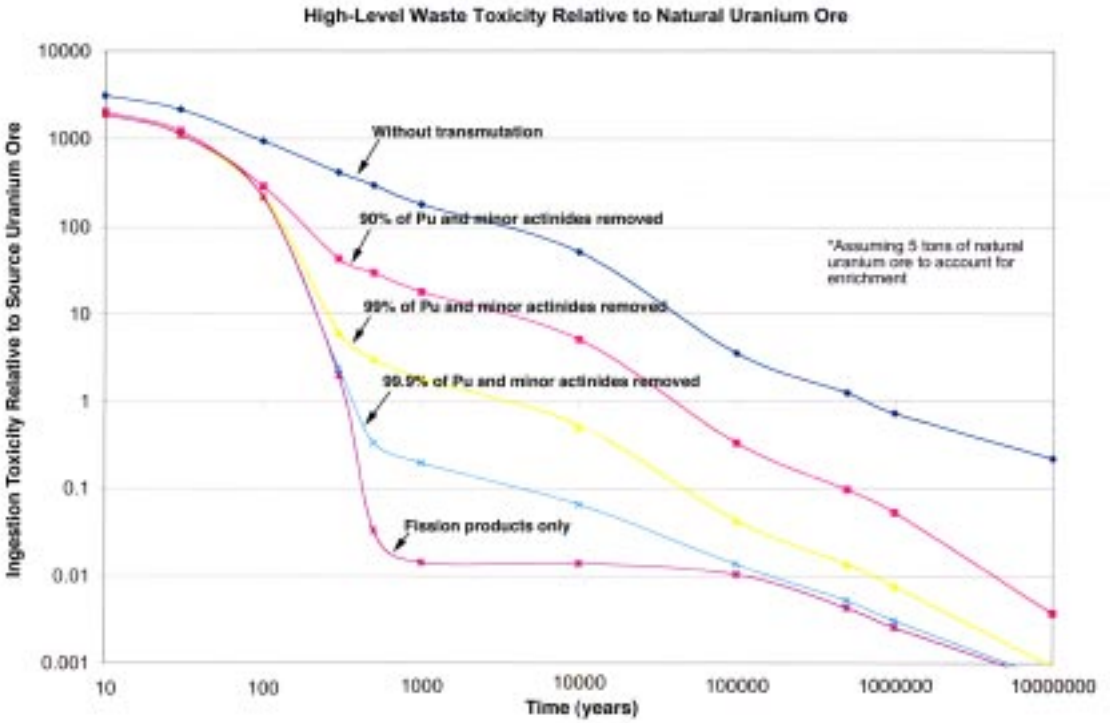


Figure 2-1. Effect of actinide removal on the radiotoxicity of spent nuclear fuel over time after removal from reactor in comparison to source uranium ore as a function of transuranic removal.

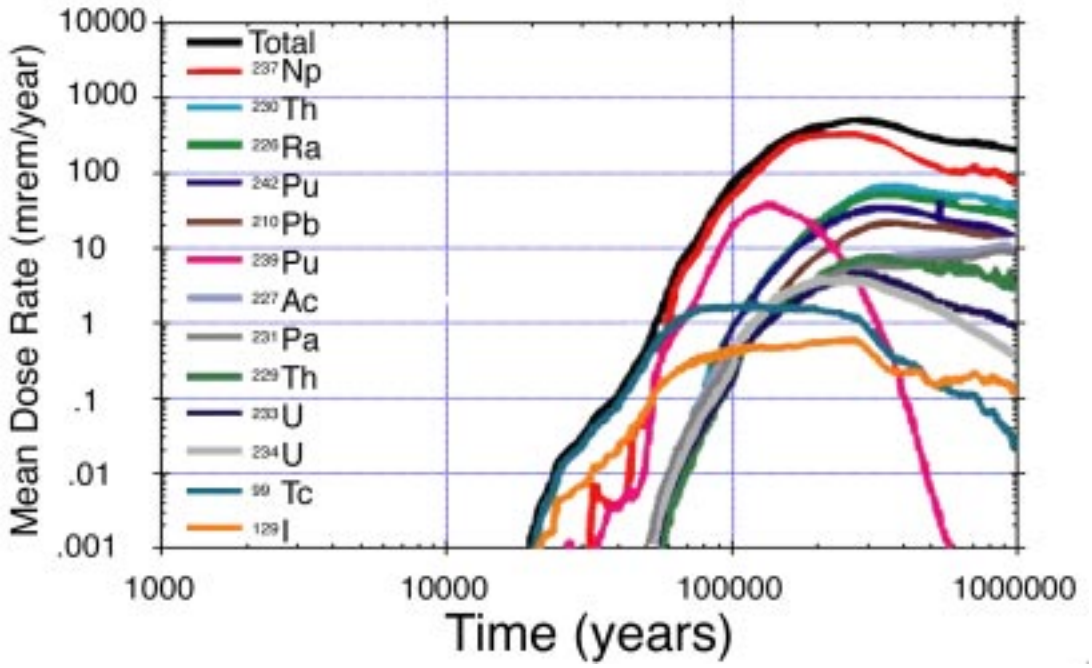


Figure 2-2. Major radioisotopic contributors to predicted dose to future inhabitants of proposed Yucca Mountain repository region.

The goal of reducing the proliferation risk from plutonium in commercial spent fuel leads to three programmatic criteria: reducing the total inventory of plutonium, reducing the amount going into repositories, and minimizing the risk of diversion while being processed in a transmutation system.

Resolution of the nuclear waste issue could improve prospects for nuclear power. Major factors, technical risk of the enterprise, and risk of increased environmental, safety, or health concerns associated with the approach taken. In principle, these criteria can be met, but they could be key discriminators in evaluating transmutation system technology options. The technical risk in pursuing integrated waste management technologies must be maintained low, since failure of the endeavor could jeopardize commercial power prospects. Another important factor, economic viability, is difficult to assess. A potential indicator of viability is the support ratio between power plants and dedicated transmutation plants, based on the assumption that the latter units are less cost effective.

Although the radiotoxicity goal is both the most visible programmatic criterion and perhaps the most limiting, it is possible that approaches that fail to meet that criterion could provide the best overall system performance, especially in light of viability and cost criteria. Therefore, some flexibility on numerical criteria may be prudent while candidate approaches are under evaluation.

The programmatic goals, metrics, and criteria are summarized in Table 2-1. In addition, the motivation or rationale for including such a criterion is provided in the right-hand column of the table.

Table 2-1. Top-Level Programmatic Goals, Criteria, and Metrics, and Motivation/Rationale for the Criteria

Top-Level Goals	Criteria	Metrics	Options for Meeting Criteria
I. Improve public safety.	I.1 Radiotoxicity Criterion: Reduce radiotoxicity of commercial spent fuel.	I.1.1 Reduce radiotoxicity of spent nuclear fuel below that of source uranium ore within a few thousand years. [Note: For this study, 1,000 years was used.]	I.1.1.1 Transmute about 99.5% of the transuranics by minimizing separations and fuel fab losses.
	I.2 Dose Criterion: Reduce radiation dose to future inhabitants of repository region.	I.2.1 Reduce maximum predicted dose to future inhabitants by at least 99% as compared to current predictions.	I.2.1.1 Transmute most neptunium, some technetium, and perhaps iodine. Place remaining inventories in superior waste forms.
II. Provide benefits to the repository program.	II.1 Heat-Load Criterion: Reduce inventory of materials that create long-term heat loads in repositories.	II.1.1 Reduce long-term heat load of spent nuclear fuel by at least 90% after 500 years as compared to unprocessed spent fuel.	II.1.1.1 Transmute 99%+ of transuranics. Evaluate separation of cesium and strontium for special packaging and handling (short-term heat load).
	II.2 Criticality Criterion: Effectively preclude future criticalities.	II.2.1 Maximize fissile material removal. Reduce inventory of transuranics in spent fuel by 99% and decrease fissile fraction in remaining transuranics.	II.2.1.1 Reduce fissile material fraction in waste repositories, especially by reducing and degrading transuranic content.
	II.3 Mass Criterion: Reduce mass requiring disposal in repository.	II.3.1 Minimize mass and volume to repository. Quantitative measure is the % reduction in mass and volume compared to once-through spent fuel.	II.3.1.1 Separate and divert uranium content. Pursue waste streams and forms that minimize mass or volume requiring deep geologic disposal.

Top-Level Goals	Criteria	Metrics	Options for Meeting Criteria
III. Reduce the proliferation risk from plutonium in commercial reactor spent fuel.	III.1 Plutonium Inventory Criterion: Reduce inventory of plutonium within fuel cycle	III.1.1 Reduce or potentially reverse the build-up of plutonium - Quantitative measure is the rate/time to reduce (by fission) 1000 kg to 1 kg (if necessary).	III.1.1.1 Transmutation system must be sufficient to overcome plutonium build-up from once through cycle.
	III.2 Plutonium Disposal Criterion: Minimize mass of plutonium transferred into repository	III.2.1 Reduce inventory of plutonium in spent fuel by 99%.	III.2.1.1 Transmute high fraction of transuranics. The plutonium fissile fraction should deplete quickly.
	III.3 Plutonium Accessibility Criterion: Minimize potential for diversion of plutonium.	III.3.1 Minimize transmutation facilities' footprint. Maximize radiation barriers with plutonium.	III.3.1.1 Keep radiation barrier isotopes with plutonium that are difficult to separate. Minimize the infrastructure and transportation.
IV. Improve Prospects for Nuclear Power.	IV.1 Viability Criterion: Provide viable and economically feasible waste management options for commercial spent fuel.	IV.1.1 Safely minimize transmuter support ratio, defined as the ratio fission rate (MWt) in transmuters to fission rate in conventional power generators. Also minimize time to earliest system deployment.	IV.1.1.1 Multi-tier systems with Tier 1 systems that fission larger fractions of transuranics; should be more cost-effective than those that pass most plutonium to Tier 2.
	IV.2 Technical Risk Criterion: Minimize technical risk to achieve solutions to nuclear waste challenges.	IV.2.1 Formally evaluate technical risk for all program and project aspects, including R&D. Minimize time, cost, and contingency required for implementation. Quantitative measure depends on methodology.	IV.2.1.1 Use international collaborations to optimize fuels and separations developments. Use pilot projects based on sound R&D to evaluate production scale. Use known technologies.
September 23, 2001	IV.3 ES&H Criterion: Improve upon ES&H characteristics of once-through fuel cycle.	IV.3.1 Minimize worker doses, waste streams, materials transport, and probabilities and consequences of postulated accidents.	IV.3.1.1 Use remote-handling systems and site facilities wisely, develop cleaner separations, and employ inherent and/or passive safety strategies.

3. Transmutation Approaches

As nuclear technology reached maturity in the 1970s and 1980s, the waste management issue became more central in several countries, and a significant fraction of the supporting research and development programs shifted from the energy-producing mission to the waste disposal mission. Both storage of spent nuclear fuel in underground repositories and transmutation of selected radioactive isotopes to ease the burden on the repository have been examined. Since then, many concepts have been proposed for achieving significant transmutation rates and numerous studies have been run to compare these concepts. Appendix B provides an overview of these studies. In several countries, significant dedicated research and development programs have resulted and are still underway; nevertheless, no program has yet succeeded in demonstrating complete closure of the fuel cycle. While there are significant international trends as to the choice of R&D paths, these uncertainties help explain why no single transmutation strategy has yet been selected as the most promising one. Further, national infrastructure and policy differences, along with different industrial approaches, have not furthered the creation of a broad consensus.

In this chapter we will first classify and describe the various concepts proposed for transmuting nuclear waste. Keeping in mind that the present study is limited in duration and resources, we will then describe the rationale employed to focus on a limited number of options, and we will describe these approaches in detail. The scope and criteria of the study will then be described.

3.1. Technological Options for Waste Transmutation

A broad variety of transmutation schemes has been proposed and studied by various groups. They can be classified in three categories, according to the source of the neutrons used for transmuting waste. First are the standard fissile fuel cycles, in which components of the waste become an integral part of the nuclear fuel used to generate fission neutrons and energy; second are the replacement fuel cycles, in which a new suite of fissile and fertile materials, distinct from the uranium-plutonium cycle, is introduced, and excess neutrons are used to burndown the waste; third are novel concepts using nonclassical neutron sources.

Standard fissile fuel cycles usually comprise a combination of reactor systems:

- Thermal reactors such as light-water reactors (LWRs), modular high-temperature gas reactors (MHTGRs), Pebble Bed Modular Reactors (PBMRs), Canada Deuterium Uraniums (CANDUs), etc., have been well-studied and shown to be very efficient for utilizing fissile uranium and plutonium, but they do tend to build up inventories of higher actinides. Several countries have implemented fuel cycles where plutonium separated from irradiated spent nuclear fuel is recycled once through these reactors. Each reactor type relies on specific fuel types, which can be either fertile (i.e., containing uranium-238 as well as fissile isotopes) or nonfertile (i.e., containing only plutonium and possibly higher minor actinides). In general, fertile fuels have reached a high level of technical maturity, whereas nonfertile fuels remain in early stages of development.
- Fast-spectrum systems, either critical or subcritical, have been shown to be very efficient for transmuting plutonium and minor actinides, and minimize any buildup of higher actinides. Fast reactor uranium-plutonium fuel cycles have been demonstrated, and limited demonstrations of minor actinide burning fuels have been performed [1].
- Note that other nonconventional reactor types such as molten-salt or molten-metal-fuelled reactors have also been proposed for the transmutation mission. These systems have been shown to be attractive under certain conditions, but their technological readiness level is usually

very low, and their development risk very high. We chose not to consider these systems in this initial study.

Combinations of these systems have been studied widely. The most common approaches are single- and two-tier systems (see Figure 3-1). Single-tier systems use a unique reactor type and a unique fuel technology, with multi-recycle of the isotopes to be transmuted; theoretical studies for fast-spectrum systems indicate that they will achieve very high transmutation rates without significant consequences on the practicality of the fuel cycle; equivalent studies for thermal systems have indicated that while high transmutation rates can also be achieved with multi-recycle, the buildup of higher actinides creates severe consequences for the fuel cycle. For this reason, only fast-spectrum single-tier systems were analyzed for this report. Two-tier systems usually combine a first thermal tier dedicated to burning fissile plutonium, and a second fast tier dedicated to transmuting minor actinides. (Note that more elaborate two-tier systems also use a fast reactor with multi-recycle to burn down all remaining plutonium in the first tier; we did not include this approach in the current study as it requires the development of an additional technology, and is predicated by an energy strategy, the total use of plutonium, which is not considered in the US.) These systems are considered the reference transmutation strategy in Japan and France, which both have large installed infrastructures of uranium- and plutonium-fueled LWRs, and a strong industrial infrastructure for separating plutonium from spent nuclear fuel.

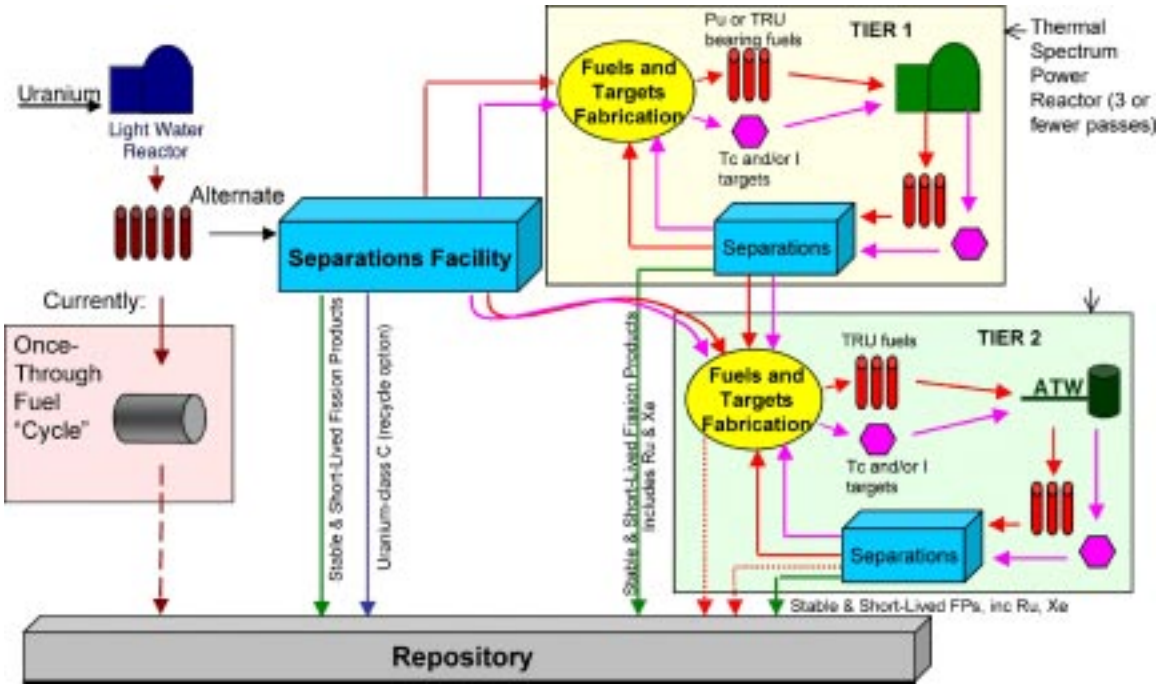


Figure 3-1. Multi-tier approach using thermal-spectrum power reactors to transmute plutonium may improve economics, but it increases materials flow complexities.

Some groups are considering multi-tier systems based on all thermal or epi-thermal systems. Because of the technical challenges related to reactivity balance and the production of higher actinides, these concepts are less well developed. Should this technology option mature somewhat, an assessment of system performance vs. goals, criteria, and metrics may be advisable.

Replacement fuel cycles are based on the fertile-fissile couple of thorium-232 and uranium-233. This cycle can be used in a sustainable, high-conversion mode, and excess neutrons used to transmute isotopes extracted from spent nuclear fuel. Various studies have indicated that this cycle does have merit. Nevertheless, the absence of a specific thorium cycle infrastructure would imply the need to develop two fuel cycle technologies, one for the thorium driver cycle and one for the transuranic burner cycle. For this reason, this approach was not considered in this initial study.

Novel concepts have also been proposed. For example, the fusion research community is proposing to use a derated fusion device for driving a subcritical molten salt reactor. Direct accelerator transmutation of waste without significant neutron multiplication has also been considered. Due to their very low degree of maturity, these concepts were not considered in this study.

3.2. Base Approaches for Task Force Evaluations

Multiple combinations of the systems described above can be envisioned. Nevertheless, the limited scope, duration, and resources available for this study constrained us to choosing a limited number of scenarios. The choice of scenarios was based on several considerations.

First, we recognized that the US situation is significantly different from that of other countries with strong transmutation programs; US policy currently discourages the separation of plutonium for civilian uses. While this situation may potentially change, it was decided to study scenarios with and without plutonium separation.

Second, we also recognized that the US has limited technical experience in transmutation technologies. This might lead to the situation where a large number of technologies look promising after preliminary inspection, but do not withstand the industrial test. Therefore, it seems reasonable to limit this study to using technologies that have received a reasonable amount of technical scrutiny here and abroad.

Third, unlike countries like Japan and France where spent nuclear fuel disposal is at least partially handled by the commercial sector, in the US this task is the sole and full responsibility of the government. Thus, one can draw an artificial boundary between the commercial sector, which creates spent nuclear fuel, and the government sector, which has the responsibility to dispose of spent nuclear fuel. Nevertheless, synergies between the two sectors are attractive; one could, for example, envision that the government could encourage the use of commercial plants to irradiate transmutation fuel. Note that while the commercial sector might be interested in using well-tested Pu-bearing fuels such as MOX, it may be reluctant to deal with americium- or curium-bearing fuels because of either the greater safety precautions required or the difficulty in handling these high-dose fuels.

Fourth, many technical studies [2, 3, 4] have indicated that there are strong advantages related to the use of fast spectra, in particular, for the effective transmutation of minor actinides. While thermal systems are, in principle, capable of transmuting all actinides, they tend to build up the higher minor actinides such as curium, berkelium, and californium, and result in very impractical fuel cycles due to the high-dose isotopes of these elements. Thus, we have systematically considered combinations of systems where the last tier is a fast-spectrum transmuter with multi-recycle of fuel.

We used the following technological building blocks:

- The spent nuclear fuel is produced by advanced light-water reactors (ALWRs) using UO_2 fuel up to a burnup of 50,000 MWd/mt. Spent nuclear fuel is cooled for 10 years before separations. The reactors are assumed to operate at constant power.
- The front-end separations are performed either using successively a combination of aqueous and dry processes to extract clean uranium and plutonium and separate the transuranic elements (TRUs) from the fission products or totally dry processes.

- The thermal tier, if considered, is made up of ALWRs with either MOX or nonfertile fuels or MHTGRs with nonfertile fuels. For either reactor type, it is assumed that the full core is loaded with these transmutation fuels. Note that while current LWRs allow only for partial MOX loadings, this limitation can easily be lifted with a new core/assembly design planned for future reactors. (The use of current LWRs with partial MOX loading is an option, but there may not be enough of these reactors available in the US to reach the desired plutonium destruction rate.) Mixed-oxide fuel uses the current French design with a maximum burnup of 53 GWd per metric ton and a single recycle (PUREX is used for that recycle). Recent French analyses indicate multi-recycle of plutonium is possible in light-water reactors, usually requiring modification to the fuel assembly design. Unfortunately, the AAA Program has very limited access to these studies, and the modeling analyses required for developing a multi-recycling strategy are very complex, well beyond the scope of the study. We plan to perform those analyses during Phase 2. Nonfertile LWR fuel uses a Swiss design, with a maximum postulated burnup of 833,000 MWd per metric ton and no recycle. MHTGR fuel is based on the TRISO concept and is not recycled. After one or two irradiations, these fuels are all recycled into the fast tier.
- The second tier is made up of either fast reactors with fertile fuel and a conversion ratio of approximately 0.5, or accelerator-driven fast systems with nonfertile fuels. In both approaches the fuel is metallic and multi-recycled using a dry pyroprocessing technology.

Note that various combinations of specific technologies can be used within each technological building block. In several approaches technological uncertainties are high, and simplifying assumptions were used, consistent with the criteria of this scoping study. For example, separations and fuel fabrication losses were assumed to be 0.1% per pass, regardless of technologies; also, all Tier 2 fuels were assumed to have similar performances. Furthermore, we do not expect significant differences in the isotopic mass flows due to variations in these technologies.

The above considerations have led us to consider the following approach configuration:

- A separate commercial nuclear power production sector is assumed; it provides a sustained constant feed of spent nuclear fuel. This sector consists uniquely of ALWRs. The fuel is UO_2 and is irradiated under the conditions described above.
- A primary discriminator for the transmutation fuel cycles hinges on US government policy concerning the allowability of plutonium separation. Another key item is the choice of thermal and fast-spectrum options (see Figure 3-1):

First strategy: two-tier systems with Pu separations

Plutonium is partially burned in the first thermal tier, and all minor actinides are burned in the fast tier. Four approaches are considered for this strategy, depending on the first-tier technology (LWR with one recycle MOX; LWR with no recycle nonfertile fuel; MHTGR with no recycle TRISO) and second-tier technologies (critical fast reactors, subcritical accelerator-driven systems). This strategy was derived from the dual-strata approach that is considered the reference case in Japan and France. Nevertheless, whereas these international programs consider a combination of thermal and fast systems to burn the plutonium and an accelerator-driven system to burn the minor actinides, the AAA Program considers the development of three distinct technologies would be too burdensome and decided to rely only on thermal reactors to burn plutonium as far as possible. This strategy is similar to Approach 2, in which transuranics are not separated.

Second strategy: two-tier systems without Pu separations

The nonseparated TRUs are first burned in the thermal tier without recycle and then further burned in the fast tier with multi-recycle. Three approaches are considered in this strategy and are similar to the

approaches for the first strategy. Options to strip out higher actinides were not considered, although removal of americium and curium would address some of the problems apparent in this approach.

Third strategy: single-tier fast systems without Pu separations

The nonseparated TRUs are burned in either an accelerator-driven system with multi-recycled nonfertile fuel or in a fast reactor with multi-recycled fertile fuel, corresponding to two approaches.

Several expert groups estimated the achievable performances for the various technologies before the start of the study. Detailed information for each approach is presented in Appendices I and J.

3.3. Scope of the Study

As described in Chapter 2, several waste management scenario studies have been analyzed in the past 20 years; these have generally been lengthy efforts run in large international contexts. The objective of this study is quite different: it was our purpose to obtain in a short time a top-level understanding of the major consequences of technology choices with respect to the ability of the various approaches to meet the criteria of the AAA Program.

Thus, the objective of this study consists of generating isotopic mass flows and waste stream flows for each scenario; these mass flows are then evaluated to estimate doses to the public and workers, radiotoxicities, fissile flows, waste volumes, and heat loads. These quantities provide the basis for comparing the performances of all approaches with respect to the AAA Program criteria.

Note that due to lack of time, resources, and agreed-upon economic models, this study does not intend to perform economic assessments of the various approaches. Furthermore, it is recognized that the tools used to generate the various results all have nonnegligible uncertainties. Consequently, unless very significant differences are observed between approaches, they will not be rigorously ranked.

The ability of the current commercial sector to provide Tier 1 plutonium destruction is difficult to access due to timing issues, i.e., the infrastructure to perform the necessary separations and fuel fabrication could not be provided soon enough to support current reactors for many years. This possibility is being assessed in a separate report.

Three fundamentally different approaches to waste transmutation, described by number of tiers and plutonium separation status (from minor actinides), were addressed in these analyses. The numbering scheme is shown in Table 3-1.

Table 3-1. Main Approach Numbering Scheme

	Pu Separated from MA	**Pu and MA Remain Together**
Two-Tier Variations	1 1X, 1Z, 1G, 1XT	2 2X, 2Z, 2G
Single-Tier Variations	N/A	3 3M, 3T

For the two-tier approaches (1 and 2), variations are labeled according to their fuel form. Those employing fertile mixed-oxide fuel are denoted with an “X.” Oxides embedded in an inert ZrO_2 matrix are denoted with a “Z.” Options employing TRISO particles are denoted with a “G,” as these are used in gas reactors. The 1XT approach differs from other Approach 1 situations in that the second tier is assumed to be a fast reactor rather than the accelerator-driven system employed in all other two-tier approaches. The single-tier approach (3) includes two fast-spectrum variations; the accelerator-driven alternative is

denoted with an “M,” while the fast-reactor approach is denoted with a “T.” These approaches are shown in relation to each other on the tree-diagram below (Figure 3-2) and are described individually in Appendix G.

Approach Tree

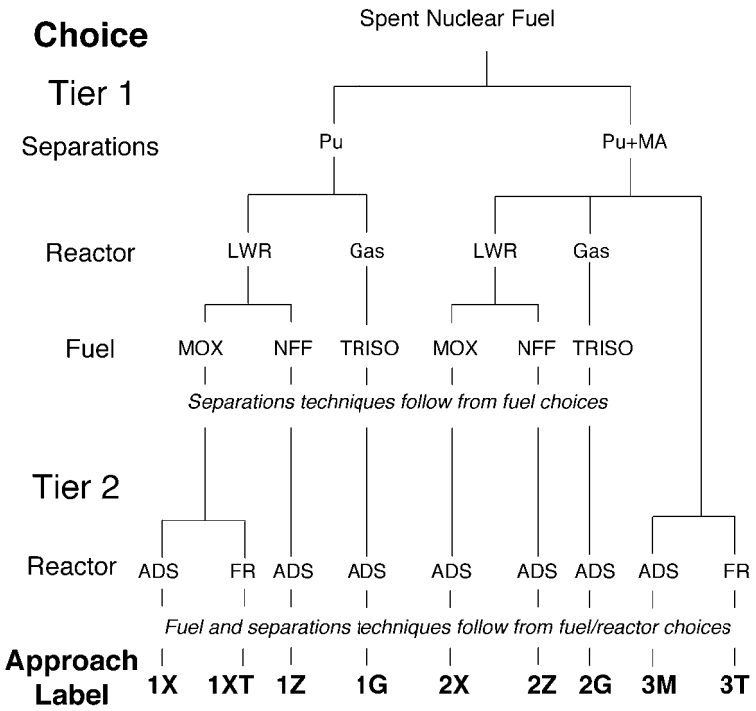


Figure 3-2. Labeling scheme for the nine approaches considered.

4. Assumptions and Methodology

4.1. Assessment Architecture

Because a prime driver has been to assess the performance of the proposed approaches against the criteria using unbiased methods, assessment architecture was used. Following the approach definition described in Chapter 3, expert judgment was solicited in the technical areas of fuels and materials, separations chemistry and reactor physics to determine not only the range of technical options and important variables, but also to determine key elements of various processes. Using the expert input with reference to previous evaluations, an integrated system process was established to serve as a flow-sheet structure for defining each of the approaches and variations. Burnup and separations calculations were performed, which allowed the system process structure to integrate mass flows and process variables through out the system. A base data set was then constructed for each approach, using system mass concentrations, thermal loading, and radioactivity for all nuclides at relevant points in the system process. Using the base data sets, and derived factors when appropriate, each approach was assessed to determine its performance against the established criteria. Specific flow sheets for each approach are provided in Appendix G.

The systematic and integrated flow of data was carefully orchestrated in this evaluation, and is depicted in Figure 4-1. To ensure material balances in the system, separations chemistry and irradiation calculations were correlated at all steps in the process. Likewise, the outputs of those calculation sets were incorporated into approach-specific spreadsheets for integration, which were then used to derive data for the criteria assessment. Data manipulation is traced in more detail in Appendix L.

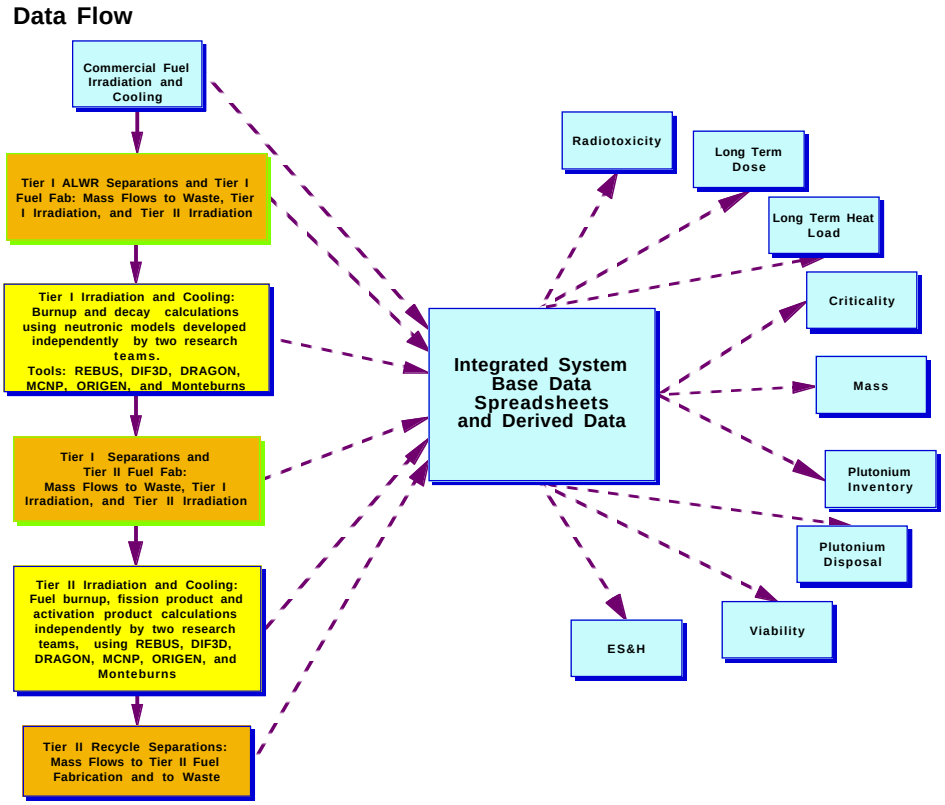


Figure 4-1. Flow of data in assessments.

4.2. Key Assessment Assumptions

Numerous assumptions were used as boundary conditions in the overall assessment, and are listed for the reader. While it is recognized that other assumptions can demonstrate variations in performance characteristics, the following set served as a ***common*** set for evaluating the approaches, and was used to ensure that the evaluation results could be compared to the criteria in an unbiased fashion. The reader is also urged to note a fundamental constraint: all evaluations were performed using ***established*** models for advanced light water reactors, gas-cooled reactors, accelerator-driven systems (ADSs) and advanced liquid-metal reactors. No system optimization was performed prior to criterion assessment, but is recommended in subsequent evaluations. The assumptions are listed relevant to their roles in the various processes.

A nuclear future exists

- Transmutation system performance is normalized to projected spent nuclear fuel input from commercial sector, *not* to commercial energy production. *Rationale: allows analysis to focus on inventory of spent nuclear fuel sent to transmutation system*
- Evaluations are normalized to electricity production in the US at same rate as year 2000, according to DOE statistics. Growth vs. decline scenarios can be assessed at a later date. *Rationale: allows analysis concentrate on performance characteristics rather than potential growth scenarios which will be assessed in the future*

Integrated system

- The commercial spent nuclear fuel feed vector is sent either to a two-tier or a one-tier system for transmutation.
- Plutonium and TRU are considered an available resource for electricity utilization in Tier 1 evaluations. *Rationale: allows analysis to assess support ratios assuming electricity production*
- Tier 1 reactors serve as reliable sources of electricity generation. *Rationale: allows analysis to assess support ratios assuming electricity production*

Commercial feed stream compositions

- The commercial spent nuclear fuel feed vector is based on the isotopic charge and discharge of a pressurized water reactor (PWR) with 50,000 MWd per metric ton burnup, as specified in *Standard- and Extended-Burnup PWR and BWR Reactor Models for ORIGEN2 Computer Code*, ORNL/TM-11018 (December 1989). *Rationale: allows common baseline for all analysis sets, feed stream isotopics are well documented, and are consistent with anticipated nuclear future*

ALWR spent nuclear fuel separation efficiency

- The commercial spent nuclear fuel is cooled 10 years before initial separations. *Rationale: allows common baseline for all analysis sets*
- The material balances were derived on the basis of 1 metric ton of TRU from the spent light-water reactor fuel. *Rationale: allows common baseline for all analysis sets, with appropriate normalization factors for case differentiation*
- U, Pu, and MA are separated at 99.9% efficiency. *Rationale: allows common baseline for all analysis sets, and insufficient data to support variable se*
- Rare earths are separated at 95% efficiency in the pyro process, and 100% removed in the aqueous process. *Rationale: process-specific characteristics*
- Separated minor actinides are sent directly to Tier 2 in approaches that utilize plutonium as the Tier 1 fuel. *Rationale: approach-specified characteristic*
- Uranium does not go to a repository for deep geologic disposition. *Rationale: consistent with Class C preference for disposition*
- Disposition of the unseparated 0.1% material is in metal and ceramic waste forms. *Rationale: consistent with current waste form technologies*
- Waste form volumes and weights are determined by the most limiting of chemical or heat dilutions necessary for disposition. *Rationale: utilizes current constraints on waste forms for assessing mass and volume criteria*

Fuel fabrication, Tiers 1 and 2

- Holdup in the fabrication is estimated at 1% to 2% in *system* equilibrium, but this is not equivalent to separation loss per pass
- Fuel fabrication losses are integrated with separations losses atom0.1%, and are not treated separately in this analysis. *Rationale: assumption is consistent with international assumptions*
- All fabrication activities require hot cell operations. *Rationale: consideration assessing sources, and for future economic analysis*

Tier 1 thermal spectrum irradiation

- All burnup calculations are based on 1 metric ton TRU. Conversion factors are used to convert masses to correspond to reductions anticipated at each cycle. *Rationale: allows common baseline for all analysis sets, with appropriate normalization factors for case differentiation.*
- Burnup calculations are constrained by a requirement to maintain criticality. *Rationale: necessary for reactor performance*
- Tier 1 light-water reactor burnup/irradiation calculations are performed with the same ORIGEN decks. *Rationale: allows common baseline for LWR analysis sets*
- Tier 1 gas-cooled reactor burnup/irradiation calculations are performed with independent MONTEBURNs, DRAGON, and ORIGEN decks. *Rationale: allows common baseline for gas analysis sets*
- Mixed-oxide cores are 100% loaded. *Rationale: allows common baseline for MOX analysis sets*

Tier 1 recycle separations

Irradiated Tier 1 Pu/MOX fuel is cooled seven years prior to recycle separation; all other irradiated Tier 1 fuel is cooled two years before separations and Tier 2 use. *Rationale: aqueous vs. pyroprocess-specific characteristic*

Tier 2 irradiation

- Equilibrium cycle employed such that Tier 1 or commercial ALWR feed is introduced at the same rate (makeup) at which the TRU is destroyed in the Tier 2 system. *Rationale: consistent with nuclear future assumption*
- All Tier 2 burnup/irradiation calculations performed with same ORIGEN decks. *Rationale: allows common baseline for all analysis sets*

Final disposition

- Waste form definition consistent with repository characteristics anticipated by Yucca Mountain geology. *Rationale: current US direction for deep geologic disposition*

As an aide to the reader, case designators and descriptions of process steps are iterated in Table 4-1. Approach-specific assumptions are provided in Table 4-2.

Table 4-1. Approach Designators

Approach Designator		Feed Stream	Pu vs. TRU	Tier 1 Fuel	Tier 1 Irradiation	Tier 2 Fuel	Tier 2 Irradiation
1X	Pu.MOX.LWR.ADS	ALWR	Pu	MOX	ALWR	Metal	ADS
1XT	Pu.MOX.LWR.ALMR	ALWR	Pu	MOX	ALWR	Metal	ALMR
1Z	Pu.NFF.LWR.ADS	ALWR	Pu	NFF	ALWR	Metal	ADS
1G	Pu.NFF.Gas.ADS	ALWR	Pu	NFF TRISO	Gas	Metal	ADS
2X	TRU.MOX.LWR.ADS	ALWR	TRU	MOX	ALWR	Metal	ADS
2Z	TRU.NFF.LWR.ADS	ALWR	TRU	NFF	ALWR	Metal	ADS
2G	TRU.NFF.Gas.ADS	ALWR	TRU	NFF TRISO	Gas	Metal	ADS
3M	TRU.ADS	ALWR	TRU	-	-	Metal	ADS
3T	TRU.ALMR	ALWR	TRU	-	-	Metal	ALMR

Table 4-2. Approach-Specific Assumptions

Applicable Approaches	Assumption
1X, 1XT	Irradiated Tier 1 Pu/MOX fuel is cooled seven years prior to recycle separation.
1Z, 1G, 2X, 2Z, 2G, 3M, 3T	All other irradiated Tier 1 and Tier 2 fuel is cooled two years before separations and Tier 2 use.
1G, 2G	Tier 1 gas-cooled reactor burnup/irradiation calculations are performed with independent MONTEBURNs, DRAGON, and ORIGEN decks.
1X, 1XT, 1Z, 2X, 2Z	Tier 1 light-water reactor burnup/irradiation calculations are performed with the same ORIGEN decks.
1Z, 2Z	Erbia (Er_2O_3) is added as a burnable poison to nonfertile fuel.
1X, 1XT, 1G, 2X, 2G, 3M, 3T	No burnable poisons were assumed for all other reactors.
1X, 1XT, 2X,	MOX-fuel assemblies are discharged at 51 GWd per metric ton.
1Z, 2Z	NFF-fuel assemblies are discharged at 510 GWd per metric ton.
1G	TRISO-fuel assemblies containing Pu are discharged at 591 GWd per metric ton.
2G	TRISO-fuel assemblies containing TRU are discharged at 470 GWd per metric ton.
1G, 2G	Gas reactors are sized at 600 MWt.
1X, 1XT, 1Z, 2X, 2Z, 3M, 3T	All other reactors are sized at 840 MWt.
1G, 2G	Tier 1 gas-cooled reactor burnup calculations must be critical (*i.e.*, $k_{eff} \geq 1$).
1X, 1XT, 1Z, 2X, 2Z, 3T	Tier 1 light-water reactor and Tier 2 fast reactor burnup calculations must be critical assuming a core neutron leakage of 3.5% Δk (*i.e.*, $k_{eff} \geq 1.035$).
1X, 1XT, 1Z, 1G, 2X, 2Z, 2G, 3M	In all Tier 2 subcritical reactors $k_{eff} = 0.97$ is targeted at BOEC.

4.3. Assumptions and Metrics in Assessing Transmutation System Performance Criteria

Only preliminary criteria assessment can be performed at this time, based on the limited data available. Table 4-3 correlates the criteria that can be assessed against the base data (mass concentration in grams, radioactivity in curies and thermal loads in watts) and derived data.

Table 4-3. Criteria Addressed vs. Metrics

Top-Level Goals	Criteria	Quantitative Criteria Calculated
I. Improve public safety	I.1 Radiotoxicity Criterion	Radiotoxicity reduction at 1,000 years
	I.2 Dose Criterion	Long-term repository dose reduction
II. Provide benefits to the repository program	II.1 Heat-Load Criterion	Heat-load reduction at 500 years
	II.2 Criticality Criterion	Pu mass reduction
	II.3 Mass Criterion	U and TRU mass reduction. Waste volume reduction
III. Reduce the proliferation risk from plutonium in commercial reactor spent fuel	III.1 Plutonium Inventory Criterion	Pu mass reduction in fuel cycle
	III.2 Plutonium Disposal Criterion	Pu mass reduction to repository
	III.3 Plutonium Accessibility Criterion	Pu mass reduction
IV. Improve prospects for nuclear power	IV.1 Viability Criterion	Support ratio
	IV.2 ES&H	Worker exposure

The assumptions and methodology for deriving data include the following:

- Waste radiotoxicity is assessed from the summation of wastes from ALWR, Tier 1 and Tier 2 separation processes. ICRP factors[3] and the summed mass are used to calculate radiotoxicity in sieverts at time of discharge. A generalized nuclide decay chain methodology was used to assess radiotoxicity up to 1,000 years, and an Approach 3M ORIGEN1 (irradiation code) run was used to extend the radiotoxicity basis for 10,000 years.
- Long-term dose was assessed using mass concentrations and toxicity information of anticipated *most-limiting* case extrapolated to remaining cases, with comprehensive calculations anticipated in the future in collaboration with the Yucca Mountain Project (YMP).
- Heat load is assessed using concentrations at discharge, which are measured against the Approach 3M ORIGEN run to evaluate heat at discharge and at 500 and 1,000 years.
- Criteria assessing Plutonium and TRU reductions are evaluated using burnup information and separation efficiency assumptions.
- Worker exposure is assessed in terms of gamma source (watts/gm $\times$ concentration in grams) and neutron source (n/s-gm $\times$ concentration in grams) during separations processes. Gamma values are derived from ORIGEN input information for the fraction of heat load due to gamma heating. Limited neutron source values are used.
- Support ratios are evaluated assuming a nuclear future that uses transmutation to handle spent nuclear fuel from the commercial sector.

[3] *The ICRP Database of Dose Coefficients: Workers and Members of the Public*, Task Force on Dose Calculations of Committee 2 of the International Commission on Radiological Protection, ISBN 0 08 042 7510.

5. Summary of Basic Mass Flow and TRU Consumption Performance

As identified in Section 3.2, three primary approaches were evaluated in this study: two-tier systems with plutonium separation, two-tier systems without plutonium separation, and single-tier fast systems without plutonium separation. These approaches will be denoted as Approaches 1, 2, and 3, respectively, in the following discussion of results. Detailed fuel-cycle flow charts for each approach are provided in Appendix G. In this chapter, the basic data of the transmutation system scoping studies are summarized; the derived parameters used to compare performance against the system criteria are discussed in Chapter 6. In addition, a more detailed evaluation of system performance is provided in Appendix H. The five distinct system types— advanced fast reactor (ALMR), commercial ALWR, transmutation ALWR, advanced gas reactor (GT-MHR), and fast-spectrum ADS (i.e., ATW)—are described in Appendix F; and the techniques used to analyze the neutron transmutation, system performance, and spent fuel characteristics are described in Appendix K.

5.1. Approach 1: Double Tier with Plutonium Separation

In this approach, elemental separation of the transuranics (i.e., to obtain a pure plutonium stream) is allowed and a thermal reactor system is used for partial burnup of the plutonium, followed by a second-tier fast-spectrum system. In this approach the minor actinides (MA) will bypass the first-tier system and proceed directly to the fast-spectrum system. For the first tier, both advanced light-water reactor and advanced gas reactor options were considered with a full-core loading of transmutation fuel. In the ALWR approach, both mixed-oxide (MOX) fuel and a nonfertile fuel (NFF) form, ZrO_2-$TRUO_2$-Er_2O_3, were considered; these approaches are denoted 1X for MOX fuel and 1Z for NFF fuel. In the MOX fuel approach, an additional single recycle of the plutonium within the first tier is employed. In the GT-MHR approach, a nonfertile TRISO particle fuel is utilized, and this approach is denoted 1G. The Tier 1 mass flow and transmutation data are summarized in Table 5-1.

Table 5-1. First-Tier Transmutation Performance for Approach 1

Parameter	Approach 1: Pu Separation, MA Bypasses Tier 1			
	1X-Stage 1	1X-Stage 2	1Z	1G
Reactor thermal power (MWt)	3000	3000	3000	600
Fuel form	MOX	MOX	NFF	TRISO
Enrichment (%Pu/HM)	8.85	13.6	100	100
Beginning-of-cycle heavy-metal (HM) inventory (metric ton)	76.9	76.9	6.38	0.61
Plutonium feed rate (kg/y)	1616	2481	1825	379
Discharge burnup (MWd/kgHM)	51	51	510	591
% TRU consumed at discharge	21.7	16.4	51.6	57.4

Note: Stage 1 is the first MOX recycle stage; Stage 2 is the second recycle stage. For Tier 1, Approaches 1X and 1XT are identical.

The nonfertile systems destroy more than 50% of the initial transuranic (TRU) mass in a single pass, whereas, the fertile MOX fuel only destroys ~20% per stage. In combination, the two-stage burnup in Approach 1X destroys one-third of the incoming TRU. Results also indicate that the second MOX stage (1X-Stage 2) requires a 50% increase in the plutonium enrichment because of the degradation of the plutonium isotopic vector; thus, mixing the first-stage discharge with the original commercial ALWR feed may be required to mitigate isotopic degradation effects on reactor control (untenable reactivity

coefficients at high plutonium content). Another striking result is the low inventory of the advanced gas reactor system (Approach 1G), only 610 kg of plutonium for a 600-MWt system. This allows the ~50% burnup to be achieved quickly (short fuel residence time).

In Approach 1, the Tier 1 discharge must be remixed with the minor actinides that bypassed the first tier for eventual transmutation in a fast-spectrum Tier 2 system with repeated recycle; for the MOX fuel approach, the MAs are also removed from the first-stage discharge and diverted to the second tier. The resulting feed compositions for the ALWR MOX, ALWR NFF, and GT-MHR scenarios are compared to the initial feed (from commercial ALWR) in Table 5-2. Roughly 50% of the TRU is passed on to the second tier for the nonfertile fuel approaches; as shown in Table 5-1 more than 50% of the plutonium was consumed, but the inclusion of the minor actinides that bypassed Tier 1 increases the overall remaining TRU content to 55%. All the first-tier approaches show a significant increase in the higher actinide content; for example, the curium-244 content increases by a factor of 3–5 as a result of the Tier 1 irradiation. Also important is the large reduction in fissile fraction resulting from the preferential consumption of these species in the thermal spectrum. The fissile fraction is roughly halved compared to commercial ALWR TRU.

Table 5-2. Second-Tier Feed Stream for Approach 1

Nuclide	Commercial ALWR Feed (w/o)	Pu Separation: MA Bypasses Tier 1		
		1X: MOX	1Z: NFF	1G: GT-MHR
Np237	6.641	7.768	6.664	6.643
Pu238	2.749	2.164	2.081	1.710
Pu239	48.652	15.054	6.092	1.821
Pu240	22.980	16.449	14.932	14.229
Pu241	6.926	7.047	7.415	5.996
Pu242	5.033	6.617	6.925	9.851
Am241	4.654	10.054	6.010	5.430
Am242m	0.019	0.051	0.029	0.021
Am243	1.472	4.152	3.134	1.671
Cm242	0.000	0.005	0.009	0.007
Cm243	0.005	0.014	0.013	0.008
Cm244	0.496	1.672	1.587	2.500
Cm245	0.038	0.212	0.185	0.138
Cm246	0.006	0.013	0.014	0.018
Total	100.000	71.273	55.193	50.100
% Fissile	55.6	31.4	24.9	15.9

Note: Feed from Approach 1X (MOX) corresponds to the second stage. For Tier 1, Approaches 1X and 1XT are identical.

For the second-tier irradiation, a fast-spectrum system with repeated recycle is utilized to destroy the remaining TRUs. The Tier 1 discharge mixed with the bypass MA (Table 5-2 composition) serves as a makeup feed to compensate for TRUs destroyed by fission. Both subcritical ATW systems using a nonfertile metal fuel and critical ALMR systems using a ternary (uranium-based) metal fuel were

evaluated; the fertile fuel ALMR system based on the MOX fuel feed is denoted Approach 1XT in the following discussion. The Tier 2 mass flow and transmutation data are summarized in Table 5-3.

Table 5-3. Second-Tier Transmutation Performance for Approach 1

Parameter	Approach 1: Pu Separation, MA Bypasses Tier 1			
	1X	1XT	1Z	1G
Reactor thermal power (MWt)	840	840	840	840
Fuel form	TRU-40Zr	U/TRU-10Zr	TRU-40Zr	TRU-40Zr
Enrichment (%TRU/HM)	99	38	99	99
Beginning-of-cycle heavy-metal inventory (metric ton)	3.40	13.8	3.66	4.07
TRU loading rate (kg/y), recycle	725	657	788	890
External makeup	**232**	143	232	232
Discharge burnup (MWd/kgHM)	223	119	208	190
% TRU consumed at discharge	24.0	17.5	22.5	20.5

The nonfertile fuel approaches achieve a high destruction rate of 232 kg/year since they do not produce any new TRU. In comparison, the fertile fuel scenario only destroys 143 kg/year, implying a conversion ratio of ~0.4. Although the low fissile content of the feed material has some performance benefits (e.g., reduced reactivity loss rate as shown in Appendix H), high TRU inventory is required to achieve the desired reactivity levels. Increased TRU inventory leads to low discharge burnup levels; a TRU consumption rate of 29% can be achieved in a single tier fast-spectrum system (see Section 5.3), this decreases to 24% in Approach 1X and 20% in Approach1G. The high TRU inventory also implies high enrichment levels for the fertile fuel approach (sized for conventional TRU feed). The production of new TRU in the fertile fuel decreases the TRU consumption rate to 17.5% per pass through the transmutation system. Thus, as expected, the fertile fuel systems require additional processing to transmute the same amount of TRU, producing proportionately more power during the transmutation process.

Technical Note: *The current commercial sector capability for the Tier 1 mission can be estimated relative to the evaluation architecture presented here by considering the scenario incorporating recovered plutonium in mixed-oxide-fueled LWRs. For this scenario the Tier 1 power requirement, assuming ALWRs, 100% MOX core loadings, and 50 GWd per metric ton of heavy metal burnup, is about 46 GWt. For a 30% core loading, this is equivalent to a required capacity of 153 GWt. Using this calculation, a rough approximation can be made of the current LWRs' capability to support this scenario. This is about half of the 300 GWt installed LWR capacity. Thus, the Tier 1 Approach 1X scenario could be supported if roughly 50% of the current LWR reactors could achieve 30% licensable MOX core loadings. This scenario provides a TRU burn of approximately 40%, but burnups near 70% may be achievable. Such deeper burn scenarios could be accommodated using more of the existing installed capacity or higher core loadings. In either approach, design modifications may drive the ability to re-license. However, the current lack of appropriate separation and fuel fabrication facilities, as well as the limited number of US reactors licensed for mixed-oxide use, renders such large-scale use of current power reactors quite hypothetical.*

5.2. Approach 2: Double Tier without Plutonium Separation

In this approach, the project assumes elemental separation of the transuranics is disallowed, but a thermal reactor system is still used for partial burnup of the TRU followed by a second-tier fast-spectrum

system. For the first tier, both ALWR and GT-MHR options were again considered with a full-core loading of transmutation fuel. In this approach, the entire load of TRU, including the minor actinides, resides in the Tier 1 fuel forms. In the ALWR approach, both the MOX and NFF fuel forms were assessed, denoted 2X for MOX fuel and 2Z for NFF fuel. In the GT-MHR approach, a nonfertile TRISO particle fuel is utilized and this approach is denoted 2G. The Tier 1 mass flow and transmutation data for Approach 2 are summarized in Table 5-4.

The fuel inventories are identical to the Approach 1 evaluation. In Approach 2Z, the same burnup was achieved by varying the burnable poison content; whereas, the GT-MHR burnup decreased slightly (to 46%) to account for the change in feed composition. Once again, the nonfertile systems destroy nearly 50% of the initial transuranic (TRU) mass in a single pass. Because the minor actinides poison the thermal spectrum, the initial enrichment is quite high (18% TRU/HM) for the MOX approach, reducing the consumption to only 14%. The high enrichment also precludes recycle stages in the first-tier MOX system and raises issues about reactor control using such high MOX fuel enrichments.

Table 5-4. First-Tier Transmutation Performance for Approach 2

Parameter	Approach 2: No Pu Separation		
	2X	2Z	2G
Reactor thermal power (MWt)	3000	3000	600
Fuel form	MOX	NFF	TRISO
Enrichment (%TRU/HM)	18.3	100	100
Beginning-of-cycle heavy-metal inventory (metric ton)	76.9	6.38	0.61
TRU feed rate (kg/y)	3332	1820	472
Discharge burnup (MWd/kgHM)	51	510	470
% TRU consumed at discharge	13.9	51.8	46.0

In this approach, the Tier 1 discharge TRU constitutes the feed material for eventual transmutation in a fast-spectrum Tier 2 system with repeated recycle. The Tier 1 discharge compositions for the ALWR MOX, ALWR NFF, and GT-MHR scenarios are compared to the initial feed (from commercial ALWR) in Table 5-5. Similar to Approach 1, roughly 50% of the TRU is passed on to the second tier for the nonfertile fuel approaches. The increases in higher actinide content are more severe than observed in Approach 1 because the MAs themselves were irradiated in the thermal spectrum; for example, the Cm-244 constitutes roughly 5% of the nonfertile approach Tier 2 feed. In addition, large reductions in the fissile fraction are observed; in the nonfertile approaches, the fissile fraction is roughly halved compared to commercial ALWR TRU. Because the burnup is low in the MOX approach, the fissile fraction is significantly higher.

Table 5-5. Second-Tier Feed Stream for Approach 2

Nuclide	Commercial ALWR Feed (w/o)	Approach 2: No Pu Separation		
		2X: MOX	2Z: NFF	2G: GT-MHR
Np237	6.641	4.342	2.405	4.202
Pu238	2.749	6.235	6.698	6.032
Pu239	48.652	32.361	4.355	4.081
Pu240	22.980	22.069	13.212	17.560
Pu241	6.926	8.119	6.653	7.137
Pu242	5.033	5.558	8.032	9.330
Am241	4.654	3.761	1.605	1.686
Am242m	0.019	0.083	0.015	0.017
Am243	1.472	1.785	2.295	0.249
Cm242	0.000	0.016	0.025	0.061
Cm243	0.005	0.020	0.031	0.028
Cm244	0.496	1.239	2.213	3.276
Cm245	0.038	0.245	0.323	0.175
Cm246	0.006	0.016	0.062	0.041
Total	100.000	85.850	48.404	54.329
% Fissile	55.6	47.6	23.6	21.1

For the second-tier irradiation, a fast-spectrum system with repeated recycle is utilized to destroy the remaining TRUs. The Tier 1 discharge (Table 5-5 composition) serves as a makeup feed to compensate for TRUs destroyed by fission. In this approach, only subcritical ATW systems using a nonfertile metal fuel were evaluated. The Tier 2 mass flow and transmutation data are summarized in Table 5-6.

Table 5-6. Second-Tier Transmutation Performance for Approach 2

Parameter	Approach 2: No Pu Separation		
	2X	2Z	2G
Reactor thermal power (MWt)	840	840	840
Fuel form	TRU-40Zr	TRU-40Zr	TRU-40Zr
Enrichment (%TRU/HM)	99	98	98
Beginning-of-cycle heavy-metal inventory (metric ton)	2.83	3.56	3.67
TRU loading rate (kg/y), recycle	585	752	779
External makeup	232	230	230
Discharge burnup (MWd/kgHM)	263	214	209
% TRU consumed at discharge	28.2	23.2	22.6

Since all approaches utilize nonfertile fuel, they achieve the maximum destruction rate of ~230 kg/year. Although the low fissile content of the feed material in the nonfertile Tier 1 approaches (2Z and 2G) has some performance benefits (e.g., reduced reactivity loss rate as shown in Appendix H), high TRU inventory is required to achieve the desired reactivity levels. Increased TRU inventory leads to lower discharge burnup levels. For example, in Approach 2X with a higher fissile content the Tier 2 burnup

increases to 28% (as compared to 23% in the other scenarios). Thus, a clear trend is observed where high burnup in the thermal Tier 1 system leads to reduced fissile content in the Tier 2 feed, which in turn reduces the achievable burnup in Tier 2. It is noted, however, that development of a nonfertile fuel containing higher actinides presents significant technical and ES&H issues.

5.3. Approach 3: Single-Tier Fast System without Plutonium Separation

In this approach, elemental separation of the transuranics is not employed, and the transmutation campaign is conducted completely in a fast-spectrum system. The *commercial* ALWR feed material (as shown in Tables 5-2 and 5-5) is fed directly into a single tier system with repeated recycle to destroy the entire inventory of TRUs. Both subcritical ATW systems using a nonfertile metal fuel and critical ALMR systems using a ternary (uranium-based) metal fuel were evaluated. The nonfertile fuel ATW approach, denoted 3M, is consistent with the ATW roadmap fuel cycle scenario evaluated in previous system studies. The fertile fuel ALMR system, denoted 3T, is similar to *actinide burning* fuel-cycle strategies postulated in previous US advanced reactor studies (i.e., Integrated Fast Reactor [IFR] program). The single tier mass flow and transmutation data are summarized in Table 5-7.

Table 5-7. Transmutation Performance for Approach 3

Parameter	Approach 3: Direct Fast System	
	3M	3T
Reactor thermal power (MWt)	840	840
Fuel form	TRU-40Zr	U/TRU-10Zr
Enrichment (%TRU/HM)	99	32
Beginning-of-cycle heavy-metal inventory (metric ton)	2.71	13.9
TRU loading rate (kg/y), recycle	553	553
External makeup	231	128
Discharge burnup (MWd/kgHM)	273	118
% TRU consumed at discharge	29.2	18.6

The nonfertile fuel approach achieves the maximum destruction rate of 231 kg/year since it does not produce any new TRU; whereas, the fertile fuel scenario only destroys 128 kg/year, implying a conversion ratio of ~0.45. The commercial ALWR feed material has a relatively high fissile content (55%), which reduces the TRU inventory in Approach 3M to only 2.7 metric tons, as compared to ~3.5 metric tons in many of the two-tier approaches. This allows the highest fast-spectrum system burnup of any of the approaches evaluated in this study, 29%. However, there are some performance penalties such as high reactivity loss rate that were observed in previous ATW studies (and shown in detailed performance comparisons in Appendix H). The high fissile content also reduced the enrichment for the fertile fuel approach to the conventional range; this results in increased TRU production (more uranium conversion to Pu-239). Thus, the makeup feed (TRU transmutation rate) is reduced to 128 kg/y, as compared to 143 kg/y in Approach 1, although the fractional TRU destruction at discharge (18.6%) is similar. Compared to the nonfertile fuel results, the fertile fuel scenario will require additional processing to transmute a given amount of TRU, but it produces more power—roughly twice the energy output—to transmute the same amount of TRU. In general, the variability in discharge burnup between the approaches implies a different number of irradiation passes to destroy the material for each scenario. The impact of this behavior on fuel-cycle losses is evaluated in Chapter 6.4

6. Comparing Performance of the Multi-Tier Approaches Against Criteria

Utilizing the base and derived data, each approach was assessed as to how it performed against the following criteria:

- Radiotoxocity reduction at 1,000 years;
- Long-term dose reduction at the repository;
- Long-term heat load at 500 and at 1,000 years;
- Plutonium reduction for criticality, inventory, and disposition;
- Mass and volume waste reduction;
- Viability through support ratio; and
- Worker exposure.

The plutonium accessibility criterion has not been addressed in this evaluation, nor has an economic assessment of the approaches been initiated.

6.1. Radiotoxicity Reduction at 1,000 Years

I.1 Radiotoxicity Criterion: Reduce radiotoxicity of spent nuclear fuel below that of the source uranium ore within a few thousand years. For this study, 1,000 years was assumed.

To calculate the long-term radiotoxicity of the spent nuclear fuel, the following assumptions were made:

- Ingestion toxicity (using the toxicity factors [in sieverts per becquerel or Sv/Bq] described in Appendix L) is a more relevant hazard to a repository than inhalation toxicity, so inhalation toxicity is ignored.
- The separations process is assumed 99.9% efficient, so that the materials going to the repository include 0.1% of all material streams (i.e., those from ALWR spent nuclear fuel separation, Tier 1 separations, and Tier 2 separations) even though the total separations efficiency for Approach 2X is only 99.8% because two separations processes are required at 99.9% efficiency each;
- The uranium separated from spent nuclear fuel has very low toxicity compared to the transuranics, so it is ignored for the figures in this section. The radiotoxicity of fission products separated from spent nuclear fuel are added to each approach because it is assumed that they are not transmuted (i.e., the potential reduction in radiotoxicity from transmutation of long-lived fission products was not addressed in this study);
- The actinides that most significantly contribute to toxicity for the first 10,000 years can be represented within several major actinide decay chains so detailed calculations on all isotopes present are not necessary; and
- The results have been appropriately normalized to system equilibrium conditions and the radiotoxicity of 7.5 metric tons of natural uranium ore (one metric ton of spent nuclear fuel is compared to 7.5 metric tons of natural uranium ore to account for the fact that 7.5 metric tons of natural uranium is required to obtain one metric ton fresh low-enriched uranium fuel for Advanced Light Water Reactors (assuming 4.2w% enrichment in U-235)).

Using the above assumptions, the toxicity resulting from each of the nine approaches was calculated relative to the toxicity of natural uranium ore at equilibrium (i.e., including all of its daughters). Figure 6-1 shows the results of these calculations for spent nuclear fuel along with the nine approaches normalized to natural uranium ore over a decay period of 10,000 years. Additionally, to see more detailed results for each approach, Figure 6-2 focuses on the timeframe between years 100 and 1,000 after the material has been separated (i.e., assuming post-cooled [typically two years] data is time 0). The approximate year that the toxicity of each case decreases to that of natural uranium ore is given in Table 6-1.

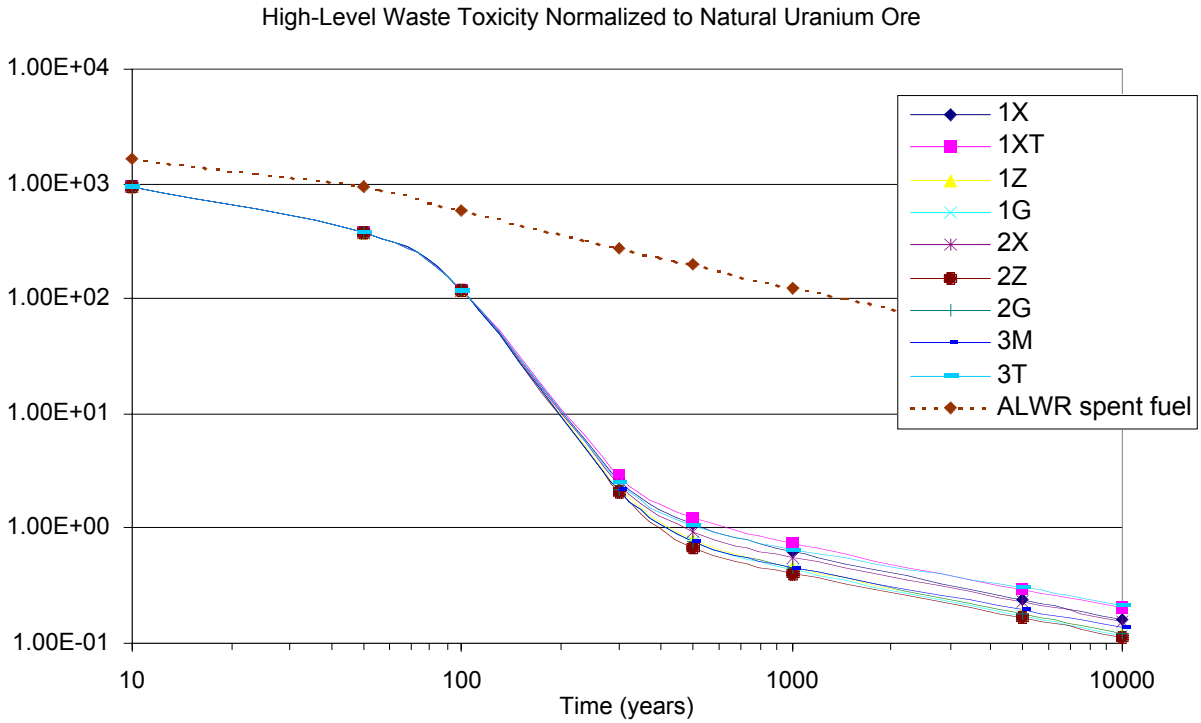


Figure 6-1. Comparison of long-term actinide toxicity of spent nuclear fuel and transmuted spent nuclear fuel to natural uranium ore.

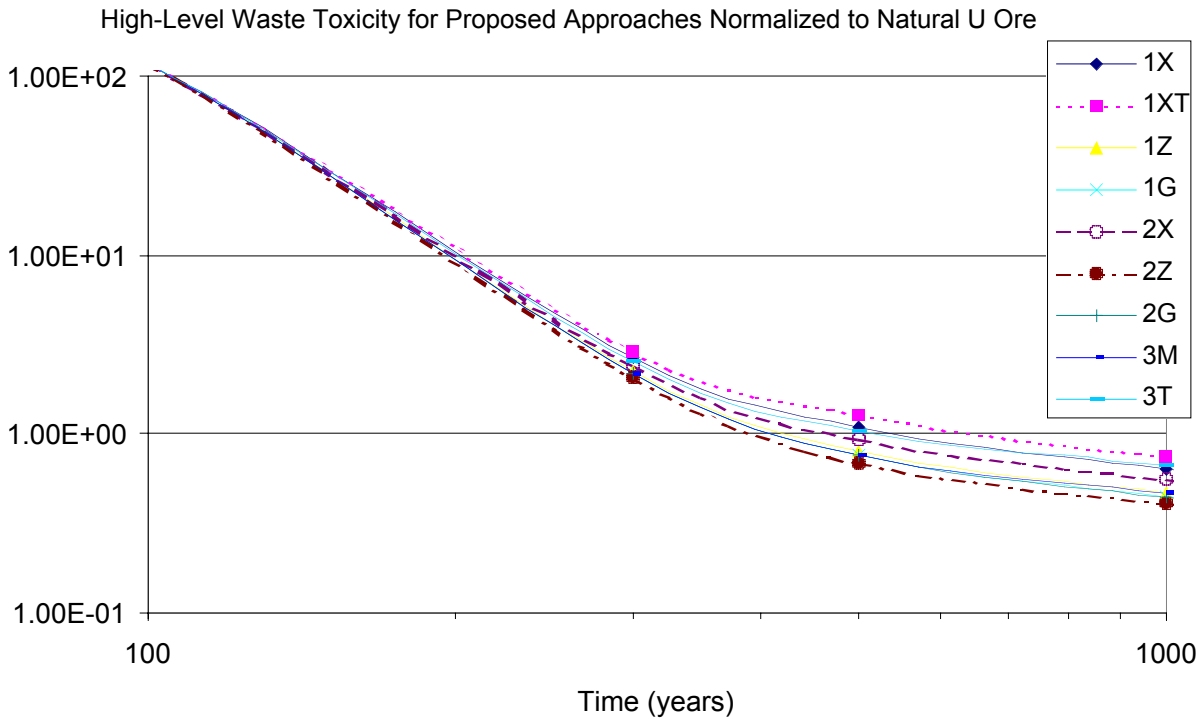


Figure 6-2. Detailed toxicity curve showing where the toxicity of transmuted spent nuclear fuel decreases to that of natural uranium ore for the various approaches examined in this study. See Table 6-1 to determine where each curve crosses 1.

Table 6-1. Number of Years Required for Transuranic Waste to Reach The Same Ingestion Toxicity as Natural Uranium Ore for Each Approach

Approach	Years to reach natural uranium level (beyond separation or post-cooling)
1X	545
1XT	670
1Z	410
1G	400
2X	460
2Z	375
2G	405
3M	395
3T	520

Observation: For all transmutation approaches evaluated in this study, the radiotoxicity is reduced below the level of natural uranium ore in less than 1,000 years.

The ingestion toxicity of ALWR spent fuel without transmutation does not become lower than that of natural uranium ore for about 300,000 years (see Figure 2-1, but keep in mind that Figure 2-1 is

normalized to 5 metric tons of natural uranium ore that has traditionally been used for spent nuclear fuel from modern LWRs as opposed to the 7.5 times enrichment used for Advanced LWRs in this study), whereas once the waste is transmuted, its toxicity will decrease to less than that of natural uranium ore in less than 1000 years, meeting the radiotoxicity criterion described above, in *all* cases. However, additional calculations show that if even a slightly less efficient separations and fabrication technology is assumed (such as 0.25%), the resulting waste will not decrease in toxicity to less than that of natural uranium ore in 1,000 years***, for any of the approaches***. Thus, separations efficiency is a key factor in decreasing long-term ingestion toxicity of spent nuclear fuel. The results indicate that using nonfertile fuel (Approaches 1Z, 1G, 2Z, 2G, and 3M) results in less repository radiotoxicity for a given ALWR spent fuel charge rate.

Another aspect of this analysis was to evaluate long-term radiotoxicity if only Tier 1 was used for transmutation and the resulting material was not separated for further transmutation but instead sent straight to a repository. Since only 13%–58% burns were obtained for Tier 1, the reduction in radiotoxicity is not significant, as can be seen in Figure 6-3 (for a comparison see Figure 2-1, which indicates what the toxicity looks like with 90% transuranic removal). In fact, toxicity was larger at many time frames for Approach 2X with only a ~14% TRU burnup because of the buildup of additional minor actinides, but even with the ~58% TRU burnup of Approach 1G, the time at which the toxicity of the remaining material decreases below that of natural uranium ore is not before 100,000 years. Therefore, performing Tier 1 operations contributes very little to meeting the radiotoxicity criterion. Also noteworthy is that after one million years, the toxicities for the Approach 2 cases are smaller than those for the Approach 1 cases because at least some of the minor actinides are burned (whereas none are burned in the Approach 1 cases).

Observation: None of the transmutation approaches can meet the 1,000-year radiotoxicity reduction criterion if the processing losses were increased to 0.25%.

Thus, low separations/fabrication losses are vital to achieving this criterion. Despite a large diversity in fuels technology between the options, 0.1% total losses assumed for all cases were applied in this study. Achieving this high efficiency will likely require a significant and development effort for any approach.

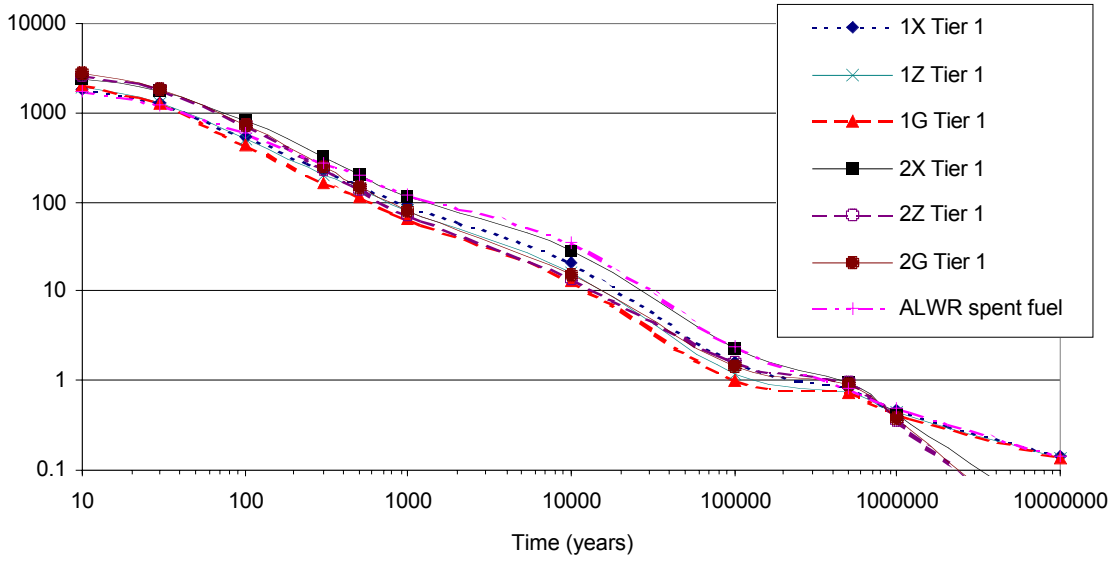


Figure 6-3. Ingestion toxicity compared to natural uranium ore if only Tier 1 is used for transmutation.

Observation: The Tier 1 thermal spectrum irradiation provides little progress toward meeting the radiotoxicity goal.

Results in Section 5 indicate the first-tier system can achieve significant plutonium consumption; however, the first-tier irradiation does not contribute significantly to radiotoxicity reduction. This study indicates that a fast-spectrum system with repeated recycle for transmuting the minor actinides is required as a final step for radiotoxicity reduction.

6.2. Long-Term Repository Dose Reduction

I.2. Dose Criterion: Reduce the maximum predicted dose to future inhabitants by at least 99% over current predictions.

Rigorous evaluation against this criterion would require a complex repository Total System Performance Assessment (TSPA) of a ***specific*** repository with and without transmutation. Because such analyses require extensive time and resources, simplified comparison to existing TSPA results are used to evaluate the currently evolving transmutation concepts. With certain assumptions about the primary contributors to repository dose, evaluation of radionuclide inventory reduction in transmutation waste streams can be used to estimate dose impact. The assumptions include:

1. The repository for transmutation wastes behaves essentially like the HLW/SNF repository concept currently under evaluation for Yucca Mountain.

This assumption is necessary to allow comparison to detailed TSPA results published for the YMP repository. Processes of importance include water flow and radionuclide transport, waste package and waste form performance, and exposure pathways.

2. Waste forms from transmutation perform as well as or better than spent nuclear fuel.

Transmutation waste forms will be different than spent nuclear fuel. However, it is likely that optimization of waste forms will result in better long-term performance.

3. For the most important radionuclides contributing to peak dose, reduction in inventory corresponds to an equivalent reduction in dose.

Under some circumstances, radionuclide mobilization is limited by solubility rather than availability, and inventory reduction may not result in equivalent dose reduction. Most of the important radionuclides appear to meet this assumption in YMP TSPA; however, uncertainty remains regarding the mobilization limits for plutonium.

With these assumptions, evaluation against the 99% reduction criterion requires the inventory reduction factor for those radionuclides capable of giving at least 1% of the dose from spent nuclear fuel. In addition, any other radionuclides that transmutation might produce enough of to become significant dose contributors must be considered. Peak dose is a sum of contributions from multiple radionuclides that vary independently with time, so not all contributing doses must be reduced by 99%, only the greatest contributors at any given time.

Examination of the YMP TSPA-SR base case shows that 11 radionuclides have individual dose peaks greater to or equal 1% of the total peak dose. Each radionuclide belongs to one of the four actinide-decay chains, and the radionuclides in each chain may be analyzed together. Only those radionuclides with half-lives long compared to fuel storage and processing times need be evaluated, as short-lived chain members will decay away during cooling (two years minimum). Similarly, very long-lived radionuclides do not contribute to dose unless present in very large quantities. This allows dose reduction to be estimated from a modest number of radionuclides.

Because Np-237 dominates the peak repository dose by about a factor of five above other contributors, the quantity of this radionuclide plus its parent nuclides Am-241 and Pu-241, must be reduced by at least 99%. Both parent nuclides have long enough half-lives to pass to disposal, but both decay almost

completely to Np-237 prior to repository peak dose. Further daughter nuclides that may also contribute significantly to dose will be reduced along with the controlling parents, and do not require evaluation unless they are produced in significant quantities during irradiation. Finally, while curium is not a major contributor to repository dose, the primary curium isotopes can increase during transmutation irradiation, and should be checked to see that they do not become major dose contributors.

Observation: Np-237 and its precursors are the dominant consideration for compliance with the peak repository dose criterion. Compliance cannot be obtained with out at least a 99% reduction in this group. If this group is reduced by more than 99%, then the criterion can be met with less reduction of other radionuclides.

The leading fission product contribution to dose comes from Tc-99. While this is dominates the early-time dose, the Tc-99 maximum is less than 1% of the total maximum by about a factor of two. Even without transmutation and somewhat more Tc-99 sent to waste from the additional fission the peak fission product dose will remain below the 99% peak reduction criterion. In actuality, it is expected that either robust waste forms or partial transmutation of Tc-99 would provide reduction in fission product dose as well.

Observation: With current assumptions, compliance of the peak repository dose reduction criterion is near the threshold requiring consideration of long-lived fission products.

Perhaps the most complex case currently analyzed is Approach 1, with two stages of ALWR plutonium-only MOX in Tier 1 and LWR-ADS Tier 2. The minor actinides from the original LWR spent nuclear fuel and from Tier 1 Stage 1 are removed and diverted to Tier 2, along with the output from Tier 1 Stage 2. Two cycles of LWR MOX provides opportunity for higher actinide growth, so this has potential to challenge the dose reduction criterion.

A simplified bounding analysis of reduction factors for this case is shown in Tables 6-2 and 6-3. Note that the waste stream from each tier and stage must be normalized back to a fixed quantity (one ton initial heavy metal) of spent nuclear fuel to allow for the comparison called for by the criterion.

Table 6-2. Waste Density by Radionuclide, in g/MT Initial Spent Nuclear Fuel

Radionuclide	SNF	SNF Separation	Tier 1 Stage 1	Tier 1 Stage 2	Tier 2	Total
U-234	210	0.2	.3	0.3	0.4	1.2
Np-237	943	0.943	0.108	0.047	1.27	2.36
Pu-238	390	0.39	0.345	0.304	3.10	4.14
Pu-239	6910	6.91	3.42	2.12	2.70	15.2
Pu-240	3260	3.26	2.88	2.31	9.24	17.7
Pu-241	983	0.983	1.06	0.99	1.70	4.73
Pu-242	714	0.714	0.876	0.932	5.46	7.98
Am-241	663	0.663	0.547	0.215	2.48	3.90
Am-243	209	0.209	0.212	0.163	2.48	3.06
Cm-243	0.76	7.6E-4	6.9E-4	5.4E-4	0.017	0.019
Cm-244	70.5	0.70	0.093	0.071	2.0	2.23
Cm-245	5.39	0.005	0.0015	0.009	0.60	0.615

Table 6-3. Sum of Each Chain and Comparison to Spent Nuclear Fuel Direct Disposal

Neptunium Chain Cm-245 + Pu-241 + Am-241 + Np-237 Total to Waste – Approach 1 = 11.6 gm Total to Waste – SNF direct disposal = 2594 gm Inventory Reduction Factor: **99.5**	**Radium Chain** Pu-242 + Pu-238 + U-234 Total to Waste – Approach 1 = 98.9 gm Total to Waste – SNF direct disposal = 1314 gm Inventory Reduction Factor: **98.9**
Actinium Chain Cm-243 + Am243 + Pu-239 Total to Waste – Approach 1 = 18.3 gm Total to Waste – SNF direct disposal = 7120 gm Inventory Reduction Factor: **99.7**	**Thorium Chain** Cm-244 + Pu-240 Total to Waste – Approach 1 = 19.9 gm Total to Waste – SNF direct disposal = 3330 gm Inventory Reduction Factor: **99.4**

Notes: Evaluation by decay chain is a first order sorting of important contributors to dose for transmutation cases pending detailed TSPA evaluation. The U-235, Pa-231 portion of the actinium chain is not included, because the long U-235 half-life reduces the importance to dose in transmutation cases compared to the Am-243, Cm-243, Pu-239 portion, while in the repository base-case calculation significant low burnup fuel and HEU fuel are dose contributors but are not present in the transmutation comparison.

These reduction factors cannot be directly combined as the peak dose contributions from each chain occurs at differing times in the repository evolution. That the Radium Chain reduction factor of 98.9 does not independently meet the 99% reduction criterion is not an issue because the peak dose from this chain is already about one order of magnitude below the total peak dose. Within the assumptions of this simplified analysis, this case should achieve the 99% dose reduction criterion.

This process applied to representative sets of the evaluation cases yields the results summarized in Table 6-4. Representative cases include once-through direct disposal with no reduction, the complex two-stage Pu MOX first tier plus ADS second tier, a two-tier nonfertile fuel with gas and LWR systems, and single-tier ADS and ALMR cases without a thermal component.

Table 6-4. Long-Term Dose Reduction Factors (%): Estimated From Key Isotope Inventory Reduction

Approach	Neptunium Chain	Radium Chain	Actinium Chain	Thorium Chain
ALWR Spent Nuclear Fuel	0	0	0	0
1X - Pu.MOX.LWR.ADS	99.5	98.9	99.7	99.4
1G – Pu.NFF.GAS.ADS	99.7	99.2	99.9	99.6
2Z - TRU.NFF.LWR.ADS	99.8	99.6	99.8	99.7
3M - ADS	99.5	99.3	99.8	99.4
3T - ALMR	99.6	99.2	99.6	99.3

Note: The Neptunium Chain represents the dominant peak dose contribution in YMP-TSPA.

6.3. Long-Term Heat-Load Reduction at 500 years

II.1. Heat-Load Criterion: Reduce by 90% or more the inventory of materials that contribute to long-term heat loads in the repository.

Long-term heat-load factors (in watts) were calculated using the thermal load per gram for each isotope as obtained from the code ORIGEN2 and concentrations of the materials at 500 and 1,000 years (using the post-cooled data as time step 0). As with radiotoxicity, it was assumed that the major contributors to the heat load at 500 and 1,000 years were actinides in several key decay chains. This assumption was justified by comparing these results to a case that was decayed out for 500 years in ORIGEN2, and the sum of heat loads for key actinides were in close agreement with the total heat load obtained from all isotopes present. Again, the results were normalized to the heat load of one metric ton spent nuclear fuel and include transuranic wastes resulting from the initial separation of ALWR spent fuel, separations during and after Tier 1, and during Tier 2. The heat loads following two years of post-cooling for all isotopes (not just actinides) were also obtained from the ORIGEN2 burnup results for each case and are normalized appropriately to ALWR spent nuclear fuel. These are all listed in Table 6-5. The heat load of uranium remaining from the initial separated ALWR spent nuclear fuel was ignored in this table.

Table 6-5. Heat Load (W) as a Function of Decay/Cooling Time For the Different Approaches Processed Through Tier 2

Approach	Post-Cooled	500 years	1,000 years
1X	7491.39	0.69	0.39
1XT	7362.64	0.80	0.45
1Z	7895.33	0.51	0.28
1G	8625.94	0.48	0.27
2X	7470.73	0.59	0.33
2Z	8005.24	0.44	0.25
2G	7978.94	0.48	0.27
3M	7590.58	0.48	0.28
3T	7673.67	0.65	0.39
ALWR spent fuel	7488.00	128.60	74.46

The same information expressed in percent reduction in heat load over the ALWR spent fuel is indicated in Table 6-6. The heat load is reduced by more than 99% in 500 years, so the criterion to reduce the heat load by 90% within 500 years was definitely met; in fact, it was exceeded.

Table 6-6. Percent Reduction In Heat Load Compared To Spent Nuclear Fuel At Specified Years After Irradiation

Approach	500	1,000
1X	99.5%	99.5%
1XT	99.4%	99.4%
1Z	99.6%	99.6%
1G	99.6%	99.6%
2X	99.5%	99.6%
2Z	99.7%	99.7%
2G	99.6%	99.6%
3M	99.6%	99.6%
3T	99.5%	99.5%

Observation: Clearly, the criterion to reduce the long-term heat load to the repository by greater than 90% can be met by all approaches, with significant margin.

6.4. Mass and Waste Volume Reduction

II.3 Mass Criterion: Minimize mass and volume to the repository by separating and diverting uranium content, and by pursuing waste streams and forms that minimize mass or volume requiring deep geologic disposal.

Table 6-7 presents estimates of waste generation for the multi-tier cases that were assessed. The values reported are totals produced from the processing, burn-up and transmutation of 1 metric ton of heavy metal (MTHM) spent LWR fuel through the full cycle (Tier 1 and Tier 2). The table indicates that the mass of TRUs and fission products that must be disposed of in the repository from all multi-tier options is basically of the same order of magnitude. The results indicate that using nonfertile fuel results in less mass to waste for a given ALWR spent fuel charge rate.

It is assumed that fission product gases are included in the ALWR Spent Fuel *FPs to Waste* total, whereas fission product (FP) gases are *not* included in any of the multi-tier cases. This assumption allows for alternate disposition of fission product gases during separation processes that the ALWR case would not undergo. If fission product gases were added to the multi-tier cases, then the percentage FP mass increase would be more on the order of 30%–75%.

Table 6-7. Waste Estimates for Multi-Tier Options

Approach	TRUs to Waste (metric ton)	FPs to Waste (metric ton)	% TRU Reduction	% FP Increase
1X	6.97E-05	6.38E-02	99.5%	17.5%
1XT	8.56E-05	6.88E-02	99.4%	26.8%
1Z	5.14E-05	5.79E-02	99.7%	6.6%
1G	5.13E-05	5.78E-02	99.7%	6.4%
2X	7.33E-05	6.01E-02	99.5%	10.7%
2Z	4.58E-05	5.77E-02	99.7%	6.3%
2G	5.04E-05	5.77E-02	99.7%	6.2%
3M	5.09E-05	5.79E-02	99.7%	6.7%
3T	7.94E-05	7.42E-02	99.5%	36.7%
ALWR spent fuel	1.49E-02	5.43E-02	0.0%	0.0%

The percentage change in transuranics and fission products is graphically represented in Figure 6-4, where it is clearly seen that the reduction in TRU for all approaches is in compliance with the criterion for two-year-cooled fuel. However, the increase in fission products, sorted by case, is more apparent in MOX-fueled systems.

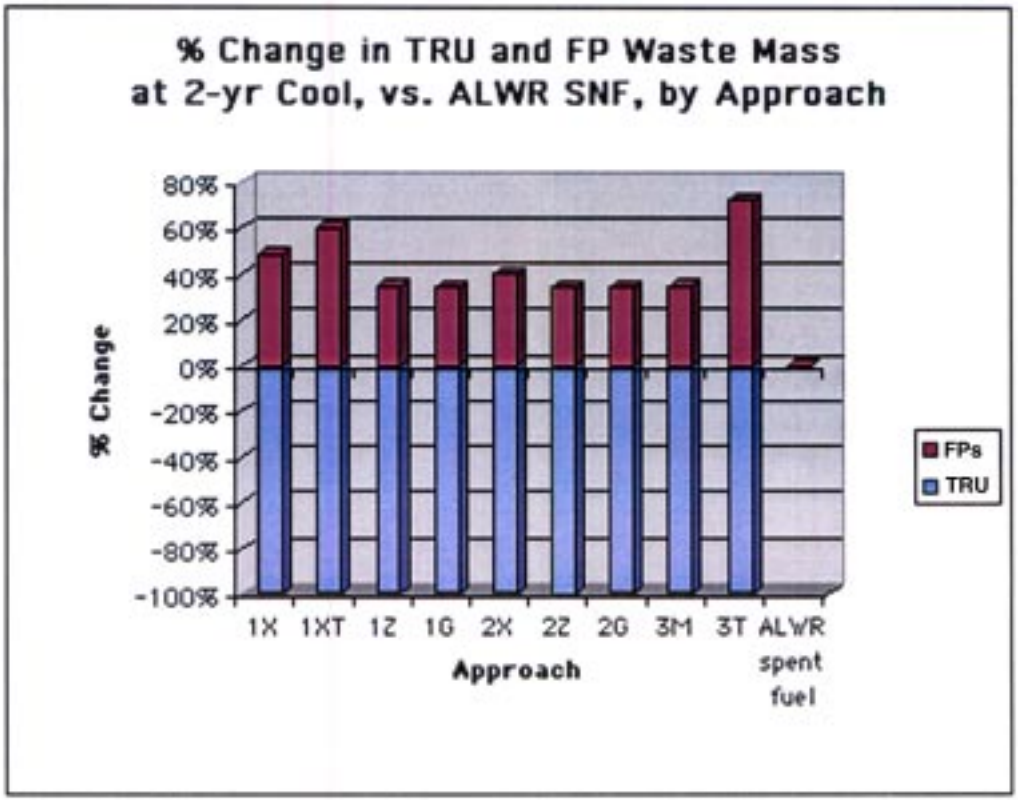


Figure 6-4. Percentage change in waste mass, assuming a two-year cooling period prior to processing.

Observation: All transmutation approaches readily achieve the 99% TRU mass reduction goal.

Observation: All approaches produce additional fission product mass (6–37%) in proportion to their power generation.

Using nonfertile fuels, the production of any new TRU is avoided resulting in the least amount of energy generation to consume the TRU; thus, they have a lower fission product waste mass. The higher FP waste mass for the fertile fuel options reflects the additional fission in the newly formed transuranics.

Observation: The fertile fuel options have slightly higher TRU losses than the nonfertile fuel approaches.

This is a consequence of the higher burnup levels obtained in the nonfertile fuel scenarios, which implies fewer processing steps (with associated losses) to destroy the material. However, it is important to recognize that these high burnups have not yet been proven for these conceptual fuel forms, whereas, conventional burnup limits were employed for the demonstrated fertile fuel forms.

In Table 6-8, waste volume at two-year cooling is explored, based on a 2000watts/cubic meter basis. The volumes were derived by assessing whether chemical dilution or thermal dilution was most dominant in the volume assessment; a description of the factors used to derive the volumes can be found in Appendix I.

Waste forms containing TRU/FP are assumed to be HLW that must be disposed in the repository. The TRUs and fission products are stabilized in waste forms including: glass, ceramic waste form, metal waste form and fluorapatite (for gas-cooled reactor fuels). Waste forms not containing TRU/FP include hardware (for ALWR fuels) and graphite materials (for gas-cooled reactor fuels) that may not require repository disposal. This determination, however, must be made with better characterization data than available for this study.

Table 6-8. Waste Volume for Multi-Tier Options, Assuming Two-Year Cooling

Approach	Waste Forms Containing TRU/FP (m^3)	Waste Forms Not Containing TRU/FP (m^3)
1X	1.95E+00	3.86E-02
1XT	1.95E+00	1.06E-01
1Z	1.77E+00	2.81E-02
1G	2.25E+00	4.09E+01
2X	1.50E+00	4.90E-02
2Z	1.39E+00	2.53E-02
2G	1.83E+00	4.09E+01
3M	1.28E+00	5.26E-02
3T	1.56E+00	1.66E-01
ALWR spent fuel	7.00E-01	

The table clearly reflects an artifact of the two-year cooling period assumption: very high heat loads at two years require that significantly higher thermal dilution factors be applied in assessing volume. Although uranium is diverted from the repository, the estimates reported in Table 6-8 suggest that a reduction in volume of waste may not be realized for any of the multi-tier options *unless cooling times are extended*. The waste volumes are dependent on the thermal power of the TRUs and fission products they contain. Existing waste forms have a limited capacity for heat. In all cases, the TRUs and fission products will need to be diluted beyond those dictated by the process to yield a waste form with acceptable heat density. Lower waste volumes than estimated could be achieved by increasing the spent fuel cool-down periods.

Observation: Thermal spectrum irradiation promotes creation of higher actinides with increased heat loading, and therefore increases the heat-limited disposition volume.

Observation: Increasing the period of post-irradiation cooling to greater than two years will allow the decay of fission products, which are a significant contributor to the thermal load and associated disposition volume.

Because of the assumption for two-year cooling, there is *very* limited cool-down data in the analytical results at this time. Using the limited data available, it has been estimated that increasing the cooling time from two to five years could decrease the heat of the fission products by approximately 20% for active metals, 85% for rare earths and 80% for noble metals, which will effectively reduce the volumes of the CWF (ceramic waste form) from Tier 2 cases to their chemical dilution levels. Corresponding Tier 1 estimates suggest that the capability to reach chemical dilution levels for glass wastes will be greater than seven years. (Note: the glass waste is a concentrated mix of all fission products, including noble metals

versus pyroprocessing where the fission products are distributed between the ceramic waste form and metal waste form and diluted more.)

The waste volumes (waste w/o TRUs and FPs) for the gas-cooled reactor cases are larger than others. A large portion of this waste consists of graphite fuel elements that hold the fuel, and may not be recyclable. General Atomics has indicated that these elements might be disposed as Class C low-level waste, but this issue is yet to be resolved.

6.5. Plutonium Mass Reduction for Criticality, Inventory, and Disposition Criteria

II.2 Criticality Criterion: Preclude possibility of future criticalities by reducing and degrading the plutonium content.

III.1 Plutonium Inventory Criterion: Reduce the inventory of plutonium in nuclear fuel cycle, reversing the long-term trend of plutonium build-up from the once-through fuel cycle.

III.2 Plutonium Disposal Criterion: Reduce the inventory of plutonium passing to the nuclear waste repository by 99% and decrease the fissile fraction within that plutonium.

Table 6-9 displays the ratio of the sum of SNF/Pu less Pu waste in Tier 1 less Pu waste in Tier 2, divided by SNF/Pu, and represents the percentage reduction of Pu sent to the repository. In all approaches, the plutonium inventory sent to the repository is reduced by greater than 99%, assuming a separations efficiency of 99.9% in an equilibrium Tier 2 cycle. Such reduction levels would satisfy the criticality and disposal criteria.

Table 6-9. Percentage Reduction of Plutonium Inventory Sent to the Repository

Approach	% Reduction
1X: Pu.MOX.LWR.ADS	99.6
1XT: Pu.MOX.LWR.ALMR	99.5
1Z: Pu.NFF.LWR.ADS	99.7
1G: Pu.NFF.Gas.ADS	99.7
2X: TRU.MOX.LWR.ADS	99.5
2Z: TRU.NFF.LWR.ADS	99.7
2G: TRU.NFF.Gas.ADS	99.7
3M: ADS	99.8
3T: ALMR	99.6

Graphically represented in Figure 6-5, the level of plutonium reduction, assuming 0.1% separations efficiency, is above 99% in all cases. This is significant when it is recalled that the ATW Requirements document stated that plutonium reduction levels exceeding 90% would certainly provide benefit to the repository, with a preference towards 99.9% in plutonium reduction.

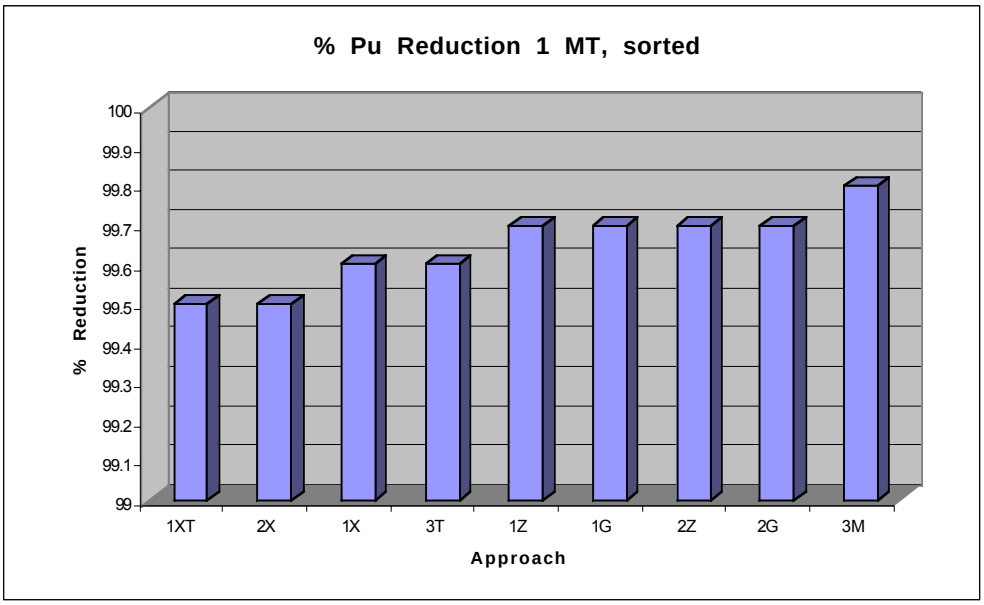


Figure 6-5. Percentage reduction of plutonium by case.

Observation: All assessed approaches exceed the criterion to reduce plutonium to the repository by greater than 99%, thereby meeting criticality and fissile material objectives as well.

With respect to the Plutonium Inventory Criterion (III.1), this criterion is not well posed with regard to an equilibrium cycle. By definition, the multi-tier approaches considered in this study will halt the plutonium buildup as soon as equilibrium is established. Evaluation of the performance of these approaches in slowing a buildup of plutonium inventory or burning down an existing inventory is highly scenario-dependent and requires detailed analyses.

Two parameters of potential interest are displayed in Table 6-10: the number of cycles required to work through (turn over) an inventory of transuranic waste and the number of cycles required to burn a specific inventory to 0.5%. The number of cycles to turn over an inventory is an indicator of the time required to make large changes in system-wide transuranic inventories. The number of cycles required to burn a specific inventory to a small residual (in this case 0.5%) is meaningful only in the unlikely event that one wishes to burn an inventory down essentially to nothing.

Because most burnup calculations performed for this study were far from optimum, the numbers of cycles shown in Table 6-10 are almost certainly overestimated. In future evaluations of multi-tier systems, these parameters can be interesting indicators of how quickly system-wide inventories can be adjusted.

Table 6-10. Number of Cycles for Inventory Turn-Over and Burn-Down

Approach	%TRU Consumed in Tier 1	# of Tier 1 Cycles	%TRU Remaining after Tier 1	%TRU Burner per Pass in Tier 2	Equilibrium Turn-Over Rate in # of Tier 2 Cycles	Total Tier 1 and Tier 2 Cycles for Turn-Over	Burn-Down Scenario: # of Tier 2 Cycles	Total Tier 1 and Tier 2 Cycles for Burn-Down
1X	28.727	2	71.273	24	3.0	5.0	18.1	20.1
1XT	28.727	2	71.273	17.5	4.1	6.1	25.8	27.8
1Z	44.807	1	55.193	22.5	2.5	3.5	18.5	19.5
1G	49.9	1	50.1	20.5	2.4	3.4	20.1	21.1
2X	14.15	1	85.85	28.2	3.0	4.0	15.5	16.5
2Z	51.596	1	48.404	23.2	2.1	3.1	17.3	18.3
2G	45.671	1	54.329	22.6	2.4	3.4	18.3	19.3
3M	0	0	100	29.2	3.4	3.4	15.3	15.3
3T	0	0	100	18.6	5.4	5.4	25.7	25.7

# of cycles = n, where (1-burnup rate)^n = 0.005.

6.6. Viability—Thermal Power Support Ratio as a Metric

IV.1 Viability Criterion: Provide a viable and economically feasible waste management option for commercial spent nuclear fuel.

One metric in evaluating the ability of a transmutation system to support or enhance the viability of nuclear power is the ability of such a system to perform this mission without dominating via large requirements for additional capacity. The thermal power support ratio is a traditional method for evaluating the capacity of a transmutation system (ADS or IFR) to provide a required material burn capability. For this evaluation, the transmutation system (Tier 2) is assumed to provide equilibrium support to burn the discharged spent-fuel TRU from a Tier 0 constant commercial reactor installed capacity of 300 GWt. This is equivalent to 255 GWt generated power at a capacity factor of 0.85. The Tier 1 system of advanced commercial reactors providing some Pu or TRU burn capability can significantly enhance the criterion Tier 2 capability and the thermal power support ratio is a good measure of this impact. Tier 2 must then support both Tier 0 and Tier 1. The thermal power support ratio is summarized in Table 6-11 for the cases evaluated. Also shown is the support ratio calculated such that the advanced reactor capacity in Tier 1 joins Tier 2 in supporting Tier 0; this is appropriate for the case in which Tier 1 capacity is assumed to exist independently.

Table 6-11. Summary of Thermal Power and Support Ratios for the Evaluation Scenarios

	Evaluation Case								
	1X Pu-ALWR-MOX-ADS	1Z Pu-ALWR-NFF-ADS	1G Pu-MHTGR-NFF-ADS	2X TRU-ALWR-MOX-ADS	2Z TRU-ALWR-NFF-ADS	2G TRU-MHTGR-NFF-ADS	3M TRU-ADS	3T TRU-IFR	1XT Pu-ALWR-MOX-IFR
	Power Generated/Required (GWt)								
Tier 0	255.0	255.0	255.0	255.0	255.0	255.0	255.0	255.0	255.0
Tier 1	54.5	31.9	30.8	20.2	37.0	28.5			54.5
Tier 2	50.8	39.4	35.7	61.4	34.7	38.7	71.6	146.3	93.3
Total – Tiers 0+1	309.5	286.9	285.8	275.2	292.0	283.5	255.0	255.0	309.5
Total – Tiers 0+1+2	360.3	326.3	321.5	336.6	326.7	322.2	326.6	401.3	402.8
	Power Fractions (%)								
Tier 1 of Tiers 0+1	17.6	11.1	10.8	7.3	12.7	10.1			17.6
Tier 2 of Tiers 0+1+2	14.1	12.1	11.1	18.2	10.6	12.0	21.9	36.5	23.2
	Support Ratio								
Tier 0 to Tiers 1+2	2.4	3.6	3.8	3.1	3.6	3.8	3.6	1.7	1.7
Tiers 0+1 to Tier 2	6.1	7.3	8.0	4.5	8.4	7.3	3.6	1.7	3.3

The support ratios presented in Table 6-11 indicate some performance trends. One is that Tier 1 provides significant material reduction capability as measured by reduced power requirements in Tier 2. An anticipated conclusion is that scenarios with fertile fuel (in Tier 1 or Tier 2) do not perform as well when judged by support ratios. Another important conclusion is that higher Tier 1 burnup scenarios (MHTGR for Pu and TRU feed) perform well and may present opportunities to achieve support ratios around 10.

Observation: In general, support ratio-based performance is best for those scenarios with nonfertile fuel in both tiers, worst for the scenarios with fertile fuel in both tiers, and somewhere in between for scenarios with fertile fuel in one tier and nonfertile fuel in the other.

Observation: The results also show variable burnup assumptions can make case-to-case support ratio comparisons difficult.

This observation is illustrated in the nonfertile fuel scenarios (ALWR and MHTGR for Pu and TRU feed) where the relative performance is dominated by the fuel burnups used. As a result, uncertainties in the maximum burnup achievable in Tier 1 impose corresponding uncertainties in support ratios.

Observation: The support ratio assessments clearly demonstrate that a significant transmutation enterprise is necessary to support a nuclear future assumption, and is variable dependent on assumptions.

6.7. ES&H and Worker Exposure

IV.2 ES&H Criterion: Improve upon ES&H characteristics of the once-through fuel cycle.

One metric for assessing the ES&H criterion is by determining potential worker exposure to neutron and gamma sources. Table 6-12 examines the strength of these sources, normalized to a metric ton (MT) of transuranics, at time periods when workers could be exposed to irradiated material: during separations and fuel fabrication processes. The data are based on factors for spontaneous fission neutron source (neutrons/sec), and gamma power (watts) derived from the ORIGEN input deck listing of the fraction of

thermal load induced by gamma heating, by isotope. As an initial screening, the approach was to apply these factors to the isotopic concentrations at any point in time following irradiation to provide an estimate for the neutron and gamma sources as a function of the material.

Table 6-12. Anticipated Gamma and Neutron Source Strengths

Sources for Worker Exposure	ALWR Seps		Tier 1 Seps Stage 1		Tier 1 Seps Stage 2		Tier 2 Seps	
	Neutron, n/s	Gamma, w	Neutron, n/s	Gamma, w	Neutron, n/s	Gamma, w	Neutron, n/s	Gamma, w
ALWR SNF	7.75E+08	4.82E+00						
1X: Pu.MOX.LWR.ADS	7.75E+08	4.82E+00	8.38E10	9.12E3	9.26E10	1.67E4	6.35E+11	7.20E+03
1XT: Pu.MOX.LWR.ALMR	7.75E+08	4.82E+00	8.38E10	9.12E3	9.26E10	1.67E4	3.50E11	5.29E3
1Z: Pu.NFF.LWR.ADS	7.75E+08	4.82E+00	1.40E+11	2.83E+04			6.80E+11	6.79E+03
1G: Pu.NFF.Gas.ADS	7.75E+08	4.82E+00	2.54E+11	2.75E+04			7.41E+11	6.22E+03
2X: TRU.MOX.LWR.ADS	7.75E+08	4.82E+00	1.39E11	1.17E4			4.59E+11	8.32E+03
2Z: TRU.NFF.LWR.ADS	7.75E+08	4.82E+00	2.53E+11	2.98E+04			8.00E+11	6.99E+03
2G: TRU.NFF.Gas.ADS	7.75E+08	4.82E+00	3.71E+11	2.10E+04			7.03E+11	6.78E+03
3M: ADS	7.75E+08	4.82E+00					3.52E+11	8.55E+03
3T: ALMR	7.75E+08	4.82E+00					1.60E11	8.50E3

Compared to all of the other processes, the ALWR separations process has the lowest gamma source. While shielding will be needed to handle these materials, the comparative dose rate should be one or more orders of magnitude less than that of other material processes.

The Tier 1 irradiated fuels during separations have a large variance in the gamma emitting isotopes of concern, that is, those that will drive the required shielding. These isotopes, Cs-134, Cs-137/Ba-137m, Rh-106/Ru-106, and Sb-125, are long enough lived to be present after the cool down period, have large activities, and gamma's of significant energy to pose shielding concerns. In order of least to most shielding, the cases are addressed in Table 6-13.

Table 6-13. Relative Gamma Source of Tier 1 Approaches

Approach	Relative Gamma Source*
Pu.MOX.LWR	1
TRU.MOX.LWR	2
TRU.NFF.Gas	25
Pu.NFF.Gas	30
Pu.NFF.LWR	30
TRU.NFF.LWR	30

*Comparison was made to Pu.MOX.LWR.

The relative difference between the MOX and the NFF fuel types is attributed to the irradiation history and cool down periods. The MOX fuel had a cool down of seven years versus the two-year period for NFF. The short-lived isotopes, Cs-134, Rh-106/Ru-106, and Sb-125 would have significantly higher activities in the nonfertile fuel by comparison. Also, the nonfertile fuels were irradiated at a higher power factor, ten times higher than the MOX fuel contributing to larger fission product inventory by comparison. Given these two differences, the relative gamma source strengths appear to be reasonable.

The variance in the gamma source for the Tier 2 irradiated fuels is significantly smaller than that found in the Tier 1 fuels. The same isotopes dominate the gamma source, Cs-134, Cs-137/Ba-137m, Rh-106/

Ru-106, and Sb-125. Compared to Tier 1, the Tier 2 fuels will result in a gamma source smaller by roughly a factor of four or five. The relative strength of the gamma sources is noted in Table 6-14.

Table 6-14. Relative Gamma Sources of Tier 2 Approaches

Approach	Relative Gamma Source*	Relative Gamma Source to Tier 1 Pu.MOX.LWR
Pu.MOX.LWR	1	2
ALMR	1.5	3
Pu.NFF.LWR	2.5	5
Pu.MOX.LWR	2.5	5
Pu.NFF.Gas	3	6
TRU.NFF.Gas	3	6
TRU.NFF.LWR	3	6
TRU.MOX.LWR	4	8
ATW	4	8

*Comparison was made to Pu.MOX.LWR.

Essentially, these fuels would fit between Tier 1 TRU.MOX.LWR and TRU.NFF.Gas.

Represented graphically in Figure 6-6, the potential for gamma exposure is more heightened in Tier 1 operations with nonfertile fuel (Pu, TRU, and TRISO) than with MOX fuel, but appear relatively the same in Tier 2. However, the fast-spectrum systems ***without*** Tier 1 demonstrate lower gamma exposure than systems ***with*** Tier 1. However, it is important to note that the dramatic gamma levels with nonfertile fuels in Tier 1 separations are because they have been cooled for only two years whereas the Pu/MOX fuels have been cooled for seven years.

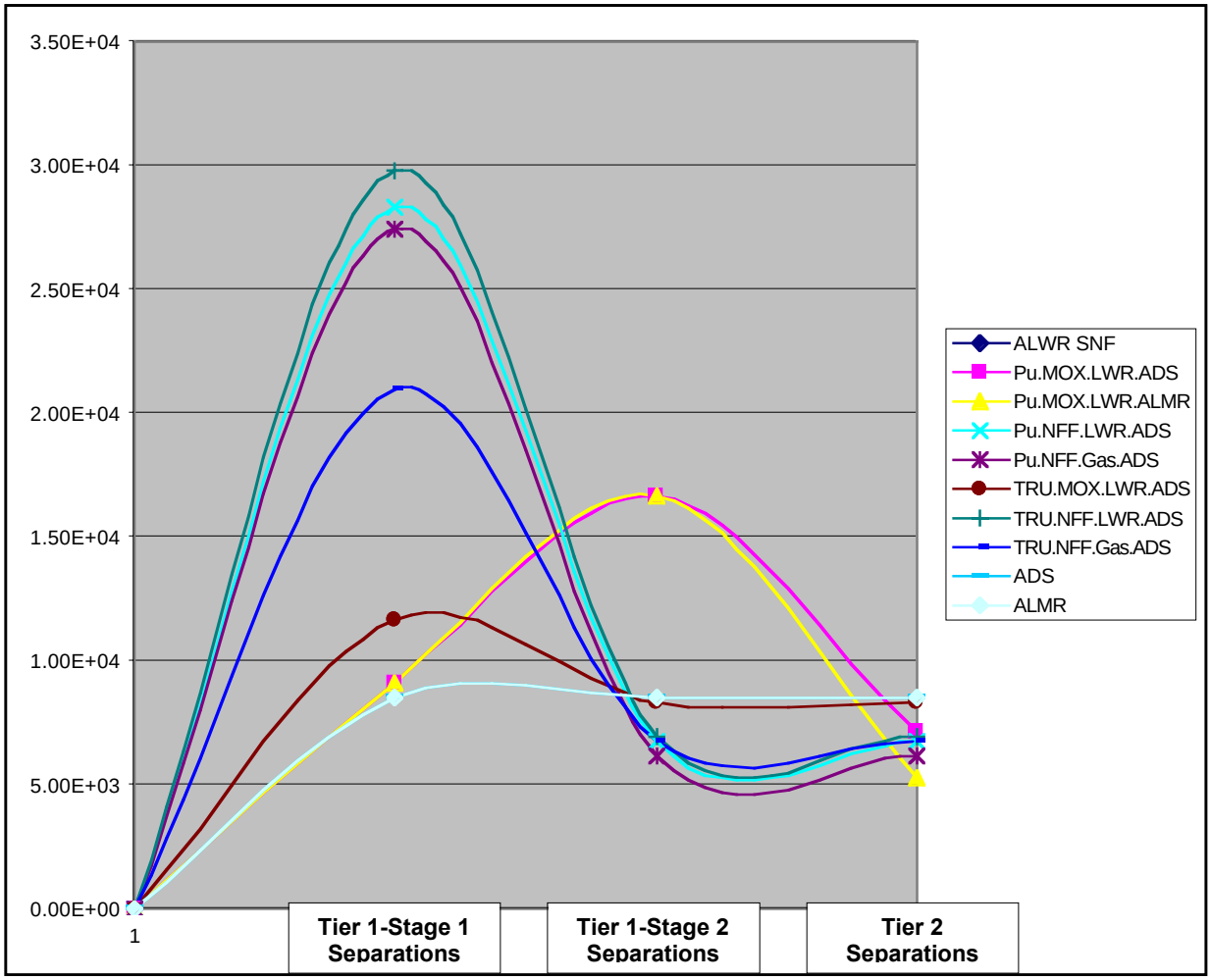


Figure 6-6. Gamma sources based on Inventory during various separations and fabrication phases.

Observation: Assessment of potential worker exposure through a quantification of neutron and gamma sources suggests that the potential for gamma exposure is more heightened in Tier 1 operations with nonfertile fuel (Pu, TRU, and TRISO) than with MOX fuel, but appears relatively the same in Tier 2. Likewise, the fast-spectrum systems without Tier 1 demonstrate lower gamma exposure than systems with Tier 1.

Clearly the actual doses would require neutron and gamma transport evaluations involving the actual materials and geometries involved in handling (processing, fabrication, etc.); this is beyond the scope of the present study. In addition, because processing will almost certainly be done remotely, estimating the dose to workers at this point is misleading. The impact of the radiation source due to various isotopic mixtures will be primarily on the costs (e.g., shielding) and time required to do the processing.

Observation: As expected, all of the irradiated fuels pose significant external exposure hazards to workers. While the differences in the relative strength appear large, there would not be large differences in the amount of shielding needed to reduce the dose rates to acceptable levels.

As a first order approximation, the addition of one to two tenth value layers of shielding material (concrete, lead) would reduce the higher gamma source fuels to an exposure level comparable to the lower source fuels. Significant work is needed to quantify the actual amount of shielding, as the

information available at the present does not lend itself to detailed shielding and worker exposure analysis. Spectra for neutron and photon emissions are needed for the next step in the analysis.

6.8. Summary of Approach Performance vs. Criteria

Contingent upon the assumed losses of 0.1% per pass, all nine proposed multi-tier approaches appear to be able to meet the programmatic criteria regarding radiotoxicity, dose, heat load, and inventory criteria. A key factor is the use of fast-spectrum, second-tier systems to fission the higher actinides.

In performing the scoping evaluations, insufficient time was available for optimization of burnup calculations, and calculations were generally halted when criticality or reactivity feedback problems were first encountered. As a result, multi-tier support ratios are most likely underestimated and the number of cycles required to process the transuranic waste stream is likely overestimated.

Efforts to estimate the resulting waste volumes and worker does must be considered preliminary. Additional cooling times would reduce waste volumes considerably and should be considered. High gamma and neutron source strengths warrant closer scrutiny, particularly in Tier 1.

7. Plans for Phase 2 Evaluations

The primary criterion for Phase 2 evaluations will to provide a basis for identifying critical issues and decision points for a long-term transmutation strategy. The evaluation will focus on fundamental technology, as well as system performance parameters. Metrics will be quantified to answer a range of questions prioritized by programmatic need. The evaluation metrics will facilitate systems trade-off assessments and support definition of program research, development, and testing strategies.

The context for the evaluation will continue to be a multi-tier architecture although lessons learned from the Phase 1 evaluations as well as input received for the Phase 1 review process will be used to refine that architecture. Similarly, the evaluation metrics used will be based on the criteria developed in Phase 1, but additional metrics will be developed as discriminators relevant to the refined architectures are identified. Identifying the impact of extending transmutation performance of Tier 1, both on the infra-structure and technologies to support Tier 1 and on the performance of Tier 2, is of particular interest.

The Phase 2 evaluation will focus on four major areas:

- Extended capability Tier 1 performance;
- Tier 1 commercial interface;
- Higher-fidelity parametric models; and
- Time-dependent performance assessment.

7.1. Extended Capability Tier 1 Performance

The transmutation capability, measured in TRU reduction, of a Tier 1 reactor is strongly dependent on the fuel-burnup and number of fuel recycles. The rationale for a deeper burn fuel cycle is based on the potential for significantly reduced material feeds to Tier 2, the resulting potential reduction in number and size of facilities in Tier 2 systems, and the higher cost of Tier 2 systems relative to Tier 1 systems. The feed to Tier 1 is determined by the spent-fuel processing assumptions and separation technologies. In Phase 1, processing was assumed to be a traditional MOX-based Pu recovery or a policy-driven decision to recover Pu with all of the minor actinides included. Since the achievable burnup is strongly dependent on the feed stock, Phase 2 will also consider an option to recover a Pu/Np product with the potential of significant material reduction without the limitations imposed on the fuel cycle by carrying the Am, Cu, and higher actinides. The focus of this evaluation element will be on the performance of feasible extensions of the material burn performance. The fuel cycles needed to address the extended capability will be treated as additional Tier 1 fuel scenarios with similar assumptions to those used in Phase 1 for system size and Tier 1/2 coupling. Although a variable feed, number of recycles and fuel burnups may be carried through the analysis, evaluations concerning feasibility and risk vs. material reduction potential will be used to select only a few deep-burn options for final two-tier assessment. In addition, the other three evaluation elements will be incorporated to provide a consistent approach across all scenarios. Primary evaluation criteria will continue to be those metrics developed in Phase 1 with explicit characterization of Tier 1 and Tier 2 contributions to the overall system performance metrics.

Phase 2 evaluations will address the separations and fuel technologies needed to support alternative feedstock and deeper-burn fuel cycles for Tier 1 reactors. In-core performance will be modeled and evaluated to obtain cell and lattice neutronic models to support credible core and fuel management concepts. Explicit models that characterize Tier 1 extended performance (e.g., burnup and recycle impacts on fuel fabrication and separations losses, discharge stream to Tier 2 systems, worker dose) will be developed. Tier 2 models will be extended where necessary to accept a range of feedstock from Tier 1

reactors. Similarly the Tier 2 separations and fuel technologies to support any qualitative differences in this feedstock will be included in the evaluation.

The adequacy of Tier 1 to address long-lived fission product (technetium and perhaps iodine) transmutation will also be included in the evaluation. Comprehensive processing flow sheets and detailed in-core performance will not be modeled. However, scoping analyses will be performed for representative concepts in order to identify the top-level issues and benefits. Material balance and losses will be assessed within the fidelity of the modeling assumptions used. Performance with respect to the criteria and metrics will be quantified where possible.

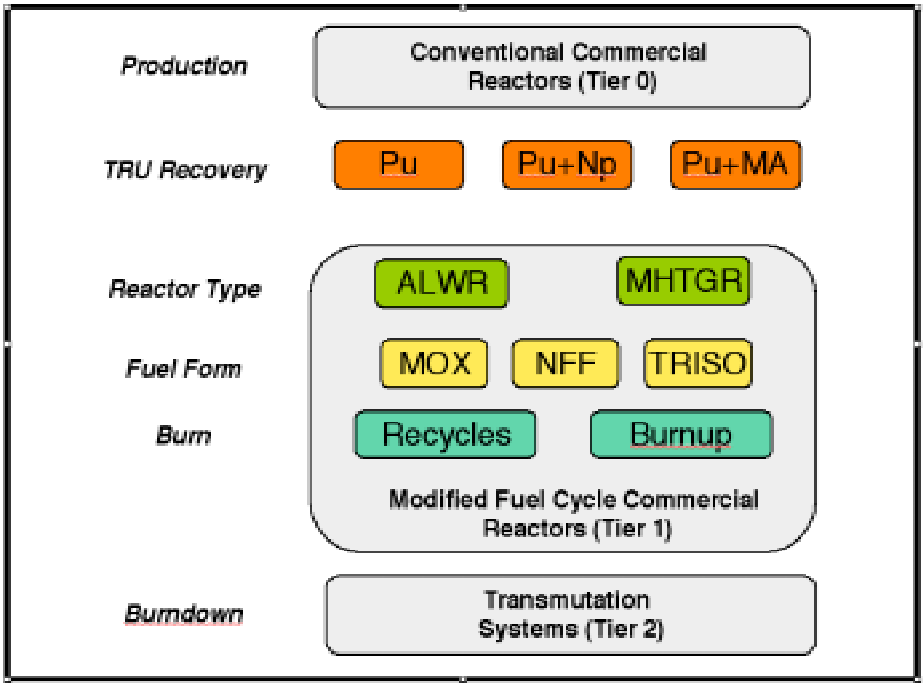


Figure 7-1. Architecture for Phase 2 evaluations.

7.2. Tier 1 Commercial Interface

A reasonable assumption is that a significant fraction of the Tier 1 mission could ultimately be performed in the commercial sector. Although the economics and risk associated with such fuel cycle options would probably not favor economic penetration into the commercial sector, a partial subsidy of the fuel cycle costs might provide significant benefit in material reduction in commercial reactors without the large reactor capital investments. The focus of this evaluation element will be on characterizing the cost and benefits of such advanced fuel cycles.

Metrics will be developed to assess the value of current and future commercial capacity to support Tier 1 fuel cycles. Parametric models will be developed based on the adequacy of advanced reactor technologies, including licensing requirements, for these systems. This evaluation element will couple strongly with the development and deployment timeline models developed for the time-dependent analyses.

7.3. Higher-Fidelity Systems Models

Identification of system performance discriminators requires increasing the fidelity of those models presently used to predict performance, followed by a thorough re-examination of transmutation scenarios modeled and analyzed in Phase 1. Increasing model fidelity is necessary to accurately differentiate the performance of the various options presently under consideration and to allow for discovery and assessment of technology and performance issues. Reasonable assumptions regarding the model fidelity and the availability of data will be made. The level of process disaggregation will be determined on the basis of current process definitions, as well as maturity. An important element of this effort will be parametric sensitivity analyses. This effort will complement the evaluation by permitting decisions about time and resource investments in data acquisition and subsystem modeling fidelity that are determined by level of performance impact. The sensitivity analyses will be useful to support any weighing of metrics as well as indicating directions for further assessment or R&D.

A major focus will be to provide multi-process models of separations and fuel fabrications operations with explicit loss characterization parameterization. This evaluation will permit a better characterization of losses in terms of location (with respect to the fuel cycle processes) within the two-tier scenarios, as well as a better assessment of multi-recycle options.

To address the issues concerning the number of recycles or maximum discharge fuel burnup in Tier 1 reactors requires the development of scaling relationships for performance vs. impact for the separations and fuel fabrication to support these fuel cycles. Impacts will include technology risk, safety, and cost. One performance metric that needs special attention is worker dose. To accurately evaluate this metric requires refined models of the radiation sources during fuel fabrication and separations. These scaling relationships will permit the selection of a few new deep-burn scenarios as described above. The results from the detailed system calculations will form the basis for an integrated systems model that tracks all performance metrics. This model will explicitly present some of the performance interdependencies. The integrated system model will include top-level costing based on the highly aggregated unit cost approach used in the Organization for Economic Cooperation and Development (OECD) study (i.e., the Delta model). Such an approach is fine for evaluating differences between thermal and fast-spectrum systems. However, to address such issues as the optimum number of recycles, the optimum burn depth may require a lower level of aggregation. Cost is a common metric that is useful both as a trade-off metric and also to assess interdependencies. Although all metrics have economic consequences, the translation of some of the metrics into cost is not straightforward. Therefore, the integrated systems model would not permit absolute systems economics to be assessed. This integrated model also serves the administrative function of tracking, archiving, and reporting provided by the spreadsheet model used in Phase 1.

Other metrics requiring model extension or enhancement include:

- Worker exposure (or exposure avoidance costs) throughout the fuel cycle, including the commercial sector (e.g., mining and milling), must be included to provide a basis for worker dose trade-offs.
- Evaluation of the possible impact of long-lived fission products, activation products, and spallation products on the waste transmutation performance; this assessment is needed to assure we are meeting the high level goals.
- Plutonium and minor actinide mass flows, inventories, and logistics to determine the extent to which intrinsic barriers exist or extrinsic barriers are needed.
- Direct functional or performance relationships between Tier 1 and Tier 2; such trade-off metrics allow a more meaningful assessment of coupling between the two tiers.

7.4. Time-Dependent Transmutation System Evaluation

Presently an equilibrium model is being used to access transmutation scenarios. Equilibrium models have several shortcomings. They do not account for the introduction of new technologies and the displacement of old technologies or the times scales associated with either of these events. In this evaluation element a dynamic model will be developed which can address the time-varying values of many of the transmutation system criteria and metrics.

In such a dynamic model, the present assumption of commercial reactors feeding Tier 1 reactors, which in turn, feed Tier 2 reactors would be based on calculated material availability as well as economic and technology readiness assessments. . Economic models developed as part of evaluation element 7.3 would be applied, as needed, to provide cost-based market penetration timelines for reactor and other technologies. Such a dynamic model would afford an evaluation of transmutation scenarios that explicitly treats the displacement of current commercial reactors and fuel cycles with Tier 1 systems. It would also treat the time-dependent issues associated with the Tier 1 and Tier 2 system deployment.

8. Summary Conclusions

The purpose of this evaluation has been to assess the performance of various multi-tier approaches to transmutation against criteria to improve long-term public safety, provide benefit to the repository, reduce proliferation potential from spent nuclear fuel and to improve the prospects of nuclear power. It was predicated on the assumption of a nuclear future, and was assessed assuming equilibrium conditions.

This preliminary evaluation assumed three primary approaches for considering multi-tier alternatives to SNF transmutation: Approach 1 assumed plutonium-fueled thermal spectrum reactors in Tier 1, with minor actinides and Tier 1 residuals going directly to Tier 2 for fast-spectrum systems to complete the transmutation; Approach 2 assumed TRU-fueled Tier 1 thermal spectrum reactors with residuals going to the Tier 2 fast-spectrum systems; Approach 3 assumed all SNF transuranics are transmuted only in fast-spectrum systems with different conversion ratios.

8.1. Principal Conclusions

At this point in the evaluation process, the following principal conclusions can be derived:

- All of the assessed approaches can fundamentally meet the transmutation criteria as stated, within the fidelity of the available data, assumptions, and analytical methods. It appears technically feasible to move toward an integrated waste management strategy using future reactors.
- Virtually every criterion, and therefore the transmutation performance of each approach, is most significantly influenced by the ability to achieve 99.9% separation of TRU materials from spent nuclear fuel.
- The current state of knowledge regarding separations and fuel fabrication losses is quite limited: while industrial scale PUREX plants have systematically achieved separation losses equal to or lower than 0.1% for uranium-oxide fuel, and to a very limited extend for mixed-oxide fuels, other processes, particularly those designed for as yet undeveloped fuel types, cannot yet be assessed with precision. Thus, detailed sensitivity studies must accompany the development of flow sheets, and a major research and development effort is required to develop and demonstrate the separations and fabrication technologies.
- Another important factor is the achievable burnup rates, which also strongly affect the overall transmutation losses (by changing the number of separations passes) and require an extensive fuel development and demonstration program.
- For two-tier systems, there is a clear advantage to trying to maximize the overall burnup rate in the first tier. As mentioned, theoretical studies indicate high potential burnup rates, but practical considerations usually limit the achievable rate. The issue needs to be studied carefully, taking into account all practical considerations in the fuel cycle.

8.2. Conclusions Against Criteria

- Each assessed approach can reduce the radiotoxicity of spent nuclear fuel to below the radiotoxicity of natural uranium ore within 1,000 years, *assuming 0.1% separation losses and the use of a fast-spectrum, second-tier transmutation system*.
- From a qualitative approach, it appears that each assessed approach can reduce maximum predicted peak dose to future inhabitants by at least 99% in comparison to current predictions.

- Each assessed approach can reduce the inventory of materials that contribute to long-term heat loads in the repository by 90% or more
- Each assessed approach reduces the transuranic mass by greater than 99%, under the given assumptions, but each cases exhibits a percentage increase in fission product mass. Likewise, there is a significant percentage increase in TRU and fission product waste volume, but this condition can be alleviated by increasing the cooling period to greater than two years. It should be noted that appropriate disposition paths for graphite fuel element material must be explored
- Each assessed approach reduces plutonium inventory by greater than 99%, which certainly exceeds the 90% nominal basis.
- The support ratio assessments clearly demonstrate a significant transmutation enterprise is needed to support a nuclear future. High support ratios could be important for making Tier 2 transmutation systems cost effective, particularly in the case of accelerator-driven systems. However, support ratio calculations in isolation from economic analyses serve only limited value in assessing each approach against the viability criterion.

8.3. Supplemental Insights from International Studies

This Multi-Tier Study focused primarily on circumstances within the US. When taking into account results summarized in the European Roadmap and in the recent Organization for Economic Cooperation and Development (OECD) study, it is possible to add the following perspectives:

- Multi-tier transmutation systems, where the maximum possible amount of plutonium is to be fissioned in thermal reactors and all minor actinides are to be transmuted in an accelerator-driven system, can be attractive. The thermal tier needs to be studied in order to reach maximum plutonium burnup (recent French studies indicate that multi-recycle of plutonium is possible in a light-water reactor). The first tier can have a very positive energy balance, feeding back into the commercial sector the energy accumulated in plutonium (Tier 1 can represent up to 25% of the energy produced in Tier 0); nevertheless, it reduces spent-fuel toxicity by a relatively limited factor (approximately 3): thus, a Tier 2 system (potentially limited to 5% of the Tier 0 park) is needed. In the case when only minor actinides are burned in Tier 2, this system must be an accelerator-driven system. The input stream to that system will be constituted only of minor actinides; this represents a major challenge for fuel fabrication and performance, and also for systems operations.
- Should plutonium separations not be allowed, recent studies published by the French CEA indicate that while multi-recycle of TRUs containing higher actinides through light-water reactors might be possible, there are severe consequences on the fuel cycle, resulting in major increases in potential worker doses. Furthermore, it is not believed that commercial operators would be tempted to handle these types of very hot fuels. Recent results published by the OECD indicate there is an economic advantage in using fertile fuels in critical reactors when both technologies and the associated separations technologies have already been researched and can take advantage of a significant technical basis.

8.4. To Do's

This evaluation has provided a set of preliminary conclusions that are strictly limited by the overriding assumptions. However, the robustness of performance in light of these assumptions and their variability, knowledge enhancement through R&D, and more comprehensive, systematic analytical approaches, must be verified in future analyses.

- Assess economic performance of approaches with attention to support ratios
- Assess the sensitivity of transmutation performance to variations in assumptions, especially the 0.1% separation efficiency assumption.
- Perform R&D on process factors that most significantly impact separation efficiency, and seek process approaches that ensure efficient separations
- Determine the maximum achievable Tier 1 burnup in balance with radiotoxicity and dose of materials sent to the repository, as well as increases in fission product inventories and waste volumes.
- Assess uncertainty incurred by lack of data, assumptions and analytical methods
- Assess alternate approaches, and variations on current approaches

9. References

Chapter 1

1. US Department of Energy Report to Congress, A Roadmap for Developing Accelerator Transmutation of Waste (ATW) Technology, DOE/RW-0519, October 1999.
2. Bowman, C.D. et al, Nuclear Energy Generation and Waste Transmutation Using Accelerator-Driven Intense Neutron Source, LA-UR-91-2601.
3. Van Tuyle, G.J. et al, "Accelerator-Driven Sub-Critical Target Concept for Transmutation of Nuclear Wastes," *Nuclear Technology*, January 1993, also BNL-NUREG-46511.
4. Chang, Y.I., and Till, C.E., "Actinide Recycle Potential in the Integral Fast Reactor (IFR) Fuel Cycle," *Transactions of the American Nuclear Society*, 1990 Winter Meeting, Washington D.C., Nov. 11-16, 1990.
5. National Research Council, *Nuclear Wastes: Technologies for Separations and Transmutation*, National Academy Press, Washington, D.C. 1996.
6. Salvatores, M., et al, "A Global Physics Approach to Transmutation of Radioactive Nuclei," *Nuclear Science and Engineering* 116, 1-18 (1994).
7. Mukaiyama, T., et al, "Importance of the Double Strata Fuel Cycle for Minor Actinide Transmutation," *Proceedings: Third OECD NEA Int. Inf. Exchange Meeting on Partitioning and Transmutation* (1994).
8. The European Technical Working Group on ADS, A European Roadmap for Developing Accelerator-driven Systems (ADS) for Nuclear Waste Incineration, April 2001.

Chapter 2

1. Minutes for the Nuclear Energy Research Advisory Committee Meeting, April 30-May1, 2001, Crystal City Marriott, Arlington, Virginia, available at hppt://www.NE.doe.gov, advisory, meetings.
2. Bowman, C.D., and Venneri, F., *Underground Autocatalytic Criticality from Plutonium and other Fissile Material*, LA-UR-94-4022, 1994.

Chapter 3

1. Baetsle, L.H. et al., *First Phase P&T Systems Study Status and Assessment Report on Actinide and Fission Product Partitioning and Transmutation* (1999), available at http://www.nea.fr/html/trw/docs/neastatus99/.
2. Wiese, H.W., "Actinide Transmutation Properties of Thermal and Fast Fission Reactors Including Multiple Recycling," Journal of Alloys and Compounds; v.271-273, pp.522-529, June 12, 1998.
3. Nifenecker, H., S. David, and O. Meplan, "Possible Use of Accelerator-Driven Subcritical Reactors for Minor Actinides Incineration (Sur l'incineration des actinides mineurs a l'aide de reacteurs hybrids), *Comptes Rendus de l'Academie des Sciences*. Serie 4, Physique, Astrophysique, no.1t.2, pp.163–284, Jan–Feb 2001.
4. Heusener, G., "Nuclear Transmutation, A Novel Waste Disposal Path? Nuclear Power Phase-Out, A Responsible Decision?" (Ist der Ausstieg aus der Kernenergie verantwortbar? Transmutation, ein neuer Weg der Entsorgung?) *Proceedings: VDI-GET Conference: Nuclear Phase-Out, a Responsible Decision?* Berlin, Germany, Sept. 28–29, 1999.

Appendix A: Nuclear Waste Stream Components and Characteristics

The blend of materials present in the spent fuel from commercial nuclear reactors is shown in Figure A-1 [1]. The primary waste component, uranium, closely approximates natural uranium and is not very toxic. Transuranics, created through neutron capture on uranium, are much more toxic than uranium or most fission products.

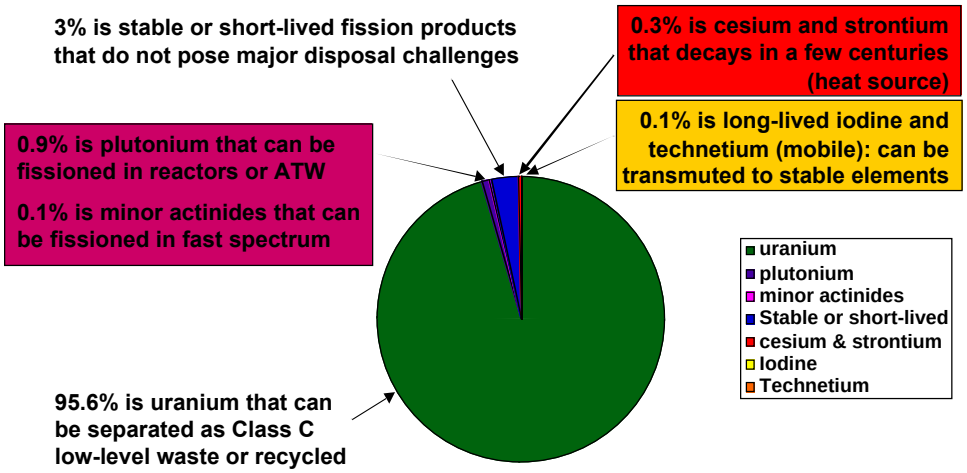


Figure A-1. Spent nuclear fuel (33,000 MWd per metric ton burnup) contains mostly uranium and stable or short-lived fission products. Most disposal issues trace to the 1% transuranic content and a handful of fission products.

Of the fission products remaining after 10 years of decay [1], only seven appear in noteworthy quantities that have either significant radiotoxicity (measured according to the amount of waste dilution required to reach drinking water standards) or very long half-lives. These are cross-compared in Figure A-2. The cesium and strontium stand out because of their high toxicity, but their short half-lives (roughly 30 years) make them short-term problems with respect to repository time-scales. They are also difficult to transmute. Of the five remaining isotopes, the technetium and iodine are potentially significant for the Yucca Mountain geology (oxidizing environment) because of their tendency to be leached out and transported by ground water.

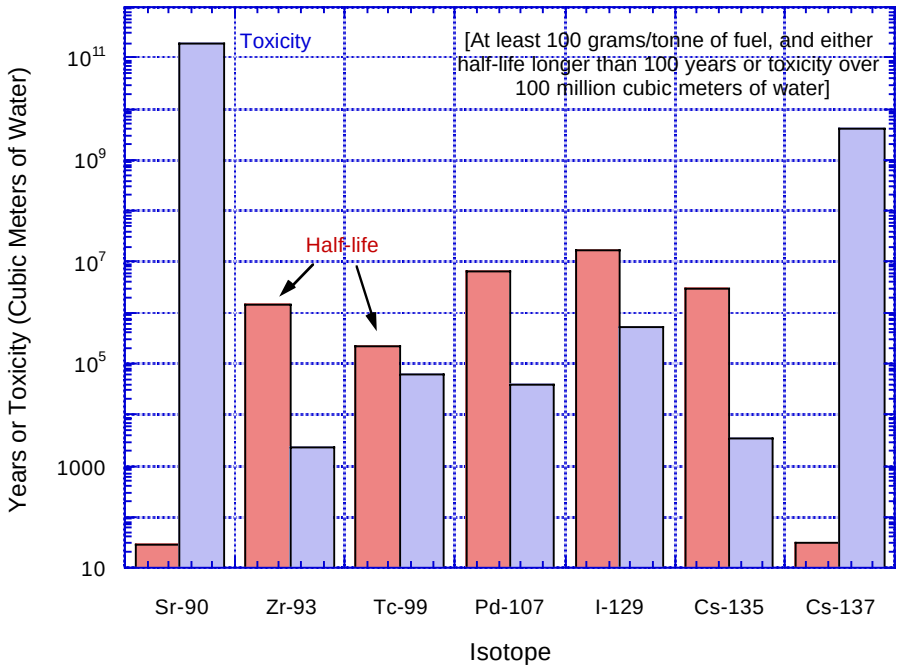


Figure A-2. For spent nuclear fuel (33,000 MWd/metric ton) fission products remaining in significant quantity after 10 years of decay, only seven fission products remain in significant quantities.

The composite toxicities in spent nuclear fuel, with and without the actinide content as it decays away over time, is shown in Figure A-3. In this approach, the ingested toxicity (again, the dilution in water to meet drinking water standards is the measure) relative to the source (for making the original fuel) uranium ore is plotted. As can be seen, nuclear waste remains more toxic than the source ore (which contains many uranium daughters and therefore significant radiotoxicity) for longer than 10,000 years. If the actinides are removed (the uranium contribution is minor—about six orders of magnitude below plutonium), the toxicity falls to that of uranium ore within about three centuries. Although not shown in this figure, removal and destruction of cesium and strontium would shift the curve further left, dropping the toxicity to uranium ore within about 30 years, but difficulties in transmuting cesium and strontium makes this impractical.

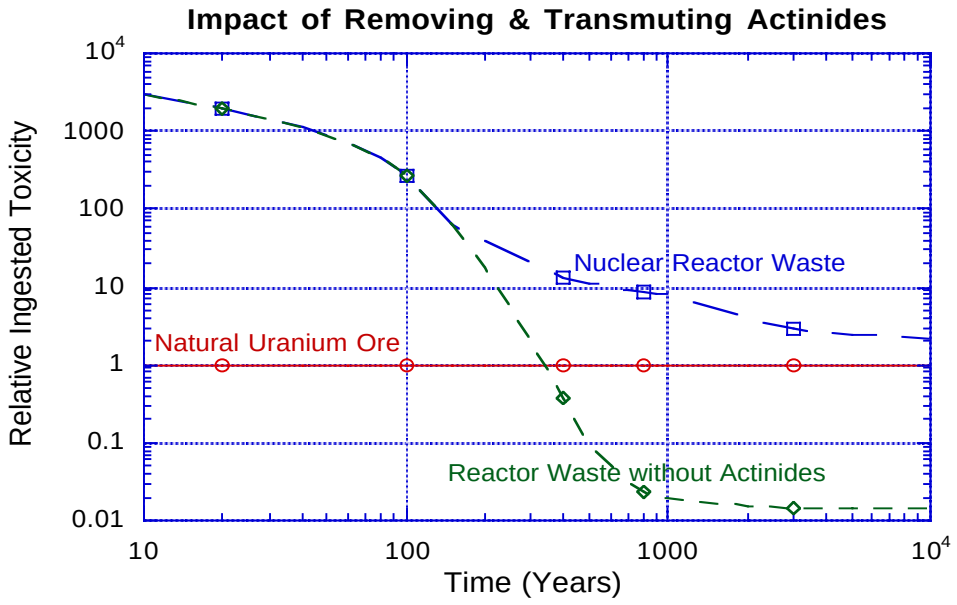


Figure A-3. Relative ingested toxicity of spent nuclear fuel, with and without the actinides, as a function of time after removal from the reactor.

Reference

1. Binney, S.E. et al., *CURE: Clean Use of Reactor Energy*, Westinghouse Hanford report WHC-EP-0268, May 1990.

Appendix B: Summary of European and Japanese Transmutation Studies Since Mid-1980s

M. Salvatores, CEA (France)

Increased interest in some countries focused attention on transmutation studies during the late eighties. In particular, Japan (the OMEGA program, which was launched in 1986-87) and France (the waste management law of 1991. The European Union has been sponsoring projects in this area since the early 1990s (3rd, 4th, and 5th Framework Programs). The Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA), in part with the support of Japan, has organized six biannual meetings on Partitioning and Transmutation since 1990 (Mito-Japan) and has produced reports on state-of-the-art technologies (*Actinide and Fission Product Partitioning and Transmutation, Status and Assessment Report*, 1999; a second report is being finalized).

The main focus of the transmutation studies in Europe and Japan have been the so-called minor actinides (MA), and to a lesser extent a few long-lived fission products (LLFP). This is because the plutonium is considered in most of these studies to be a resource, i.e., a reactor fuel.

In terms of number of studies and funds, the activities related to chemistry have been more or less 70% of the total. These studies will not be discussed here.

The performed transmutation studies can be subdivided according to:

- The recycling mode: Homogeneous (MA homogeneously mixed with the main fuel components of standard reactors) or heterogeneous (MA and/or LLFP targets at specified positions in a standard reactor).
- The type of reactor where the recycling takes place:
    - Thermal reactors (mostly PWRs, but studies have been performed for BWRs, MHTGRs, and CANDUs).
    - Fast reactors (with liquid-metal coolants, Na, Pb or Pb/Bi, or gas coolant), with different types of fuel (oxide, nitride, or metal).
- Dedicated reactors (critical or subcritical, with thermal or fast spectrum, with solid or molten salt fuels). These reactors have a nonfertile fuel. According to the chosen strategy, Pu has also been associated with MA.
- The strategy and scenarios related to the separation of Pu from MA (separation allowed or not).

Both Europe (e.g., CEA and the EU-funded “strategy” study) and Japan (JNC) have made studies with nonseparated Pu and MA; these studies were mostly based on the homogeneous recycling in fast reactors. The same institutes studied plutonium separated from minor actinides in the framework of the heterogeneous recycling mode (i.e., targets of minor actinides in specially moderated subassemblies at the periphery of a standard, MOX-fuelled, fast reactor).

The separation of Pu and MA also has been the central point of the so-called “double strata” strategy, proposed and studied first at JAERI (Japan) and further explored in France.

As far as critical reactors, the studies have covered the following fields:

- Impact of the presence of MA on the core performances (reactivity coefficients, power distributions, reactivity loss during the cycle, etc.) The studies have allowed one to define upper limits of MA concentrations in the core.

- Impact of cross sections uncertainties. These studies have triggered new experimental activities in the field of MA cross-section measurements (e.g., the N-TOF experiments at CERN-Geneva and at IRMN-Geel).
- Impact on structural material performances (e.g., maximum admissible damages on MA target-cladding in the heterogeneous recycling).
- Impact of the presence of lanthanides in the fuel.

As far as fuels, the conceptual core studies have been the basis for launching theoretical and experimental activities on fuels containing MA and on LLFP targets:

- For the homogeneous recycling in fast reactors, the SUPERFACT experiment in PHENIX has been fully analyzed. A new experiment (METAPHIX) on metal fuel containing MA is foreseen in PHENIX.
- For the heterogeneous recycling, Am-based targets have been fabricated and irradiated in the HFR reactor. Further experiments are expected in PHENIX. A substantial effort has been devoted to the choice and to the experimental validation of the “inert matrix” materials, which act as support of the actinides in the target. Both irradiations in reactors (neutron effects) and in accelerators (fission-product effects) have been used for that objective.
- For dedicated cores, theoretical studies have indicated a number of potential fuel types candidates, in particular of the CERCER and CERMET type. Fabrication/irradiation programs have been launched (in Europe, with the support of the EU).
- Tc-99 and iodine-based targets have been fabricated and irradiated (HFR).

As far as the impact of transmutation on the fuel cycle, extensive studies have been performed for most of the scenarios indicated above to quantify:

- The impact of transmutation (and partitioning) on the performance of a deep geological storage, mostly in terms of potential source of radiotoxicity reduction.
- The impact on the operations of the fuel cycle: fuel processing, fuel fabrication, etc., mostly in terms of decay heat, γ, and n doses.
- The impact of a possible long-term storage of separated Cm.

Studies have also been performed on the use of the thorium cycle as an alternative to the uranium cycle to reduce the radiotoxicity “at the source.”

As far as accelerator-driven systems, motivation studies have been performed to define the potential role for these systems (see in particular the recently released European Roadmap towards the deployment of ADS systems, and the OECD-NEA report comparing ADS and critical fast reactors as “transmuters,” being finalized now). Several experimental programs related to the validation of basic components of an accelerator-driven system have been launched in Europe:

- The IPHI and TRASCO programs for high-intensity proton accelerators.
- The MUSE experimental program on the neutronics of a subcritical source-driven core at the MASURCA installation in France.
- The MEGAPIE 1 MW pilot experiment of a full scale Pb/Bi target, to be irradiated at the PSI cyclotron in Switzerland.
- Experiments on the properties and behavior of lead alloys (the KALLA laboratory at FZK-Germany and the CIRCE loop in Italy, among others).

- Experiments on materials under neutron and proton irradiation, also with simulated spallation product implantation (e.g., for window material characterization).
- Experiments on the physics of spallation (neutron production, spallation product distributions etc.)

Regarding chemistry, as indicated above, a substantial effort has been devoted to the development of separation schemes for MA in the frame of aqueous reprocessing and the experimental demonstration of their scientific feasibility. Recently, experimental activities have been launched to investigate the potential of pyrochemical processes.

Appendix C: From "Minutes for the Nuclear Energy Research Advisory Committee Meeting"

April 30–May 1, 2001
Crystal City Marriott
Arlington, Virginia

Available at http://www.NE.doe.gov/advisory/meetings

Monday, April 30, 2001

Chairman Duderstadt called the meeting to order at 10:00 a.m. and asked each Committee member to introduce himself or herself. He reviewed the agenda and introduced Burton Richter to present the report of the ATW (Accelerator Transmutation of Waste) Subcommittee.

An initial ATW roadmap was prepared in FY99 (Roadmap I) and described the processes for treating 78,000 tons of spent fuel from the present generation of nuclear power plants. First, the spent fuel would be partitioned into three parts: U (95% by mass); Pu, Np, Am, Cm, and long-lived fission fragments; and short-lived fission products (300-year storage required). The actinides and long-lived fission products would be transmuted by the use of special-purpose, large (800-MWt) subcritical reactors driven by high-power accelerators. The accelerator design was based on technology developed for the accelerator production of tritium. The job would take more than 100 years to treat the spent fuel.

The Subcommittee developed four criteria for judging a transmutation program:

- Reduce the long-term radiotoxicity.
- Reduce the long-term proliferation risk.
- Benefit to the repository storage.
- Increase the acceptability of nuclear power (including economics).

Criteria 1 to 3 can be evaluated relatively soon, but Criterion 4 (economics) will require much longer.

There are many approaches to designing a transmutation system; choosing the right approach is important. The Subcommittee believes that the assessment of this technology should proceed on a modified basic assumption, that nuclear power will be a long-term part of the US energy program. Also, the assessment should examine multiple approaches and configurations, including combinations of critical and accelerator-driven systems, such as

- A once-through reactor plus a large accelerator-driven system (ADS),
- A light-water reactor (LWR) plus a smaller ADS,
- An LWR plus mixed-oxide fuel (MOX) plus an ADS (a fuel system used by the Europeans),
- A fast-spectrum reactor acting alone, and
- Variants on all of the above.

The Subcommittee has recommended that Option 1 be dropped; the Office of Nuclear Energy Research (NE) agrees.

He sketched out the classic approach (Option 1, above) and the dual-strata schema (which recycles some of the spent fuel to the reactor to diminish the amount of material that must be sent to the

accelerator). This cycling can be done again and again to reduce the amount of material that must be treated in the accelerator and the amount that must ultimately go to a repository. The potential payoffs of such a design are:

- The long-term radiotoxicity is reduced to below that of uranium ore in about 3,000 years with once through. (Note that a repository is still required.) No detailed design of such a system is available, but the operation of French and British reprocessing plants indicates that such plants have a negligible radio logic effect on the general public.
- Proliferation potential is reduced because, although a fixed level of nuclear power gives a linear Pu buildup over time, transmutation gives a constant level equal to the “in process” Pu. Several cycles of Pu burning are required, but the isotopic mix after the first pass through makes weapons building very difficult. [This question was posed to the National Nuclear Security Administration (NNSA) for analysis.] You could make a weapon out of this in principle, but that is not likely in practice.
- The mass of the actinides and fission products in the repository is reduced by a factor of about 20 (to 5% of the original). The volume is reduced by a factor of 3 to 10. The shorter required storage time simplifies design of containers and the repository itself.
- The impact on nuclear-power economics is unknown now and cannot be determined until the R&D is further along. (Because about 25% of the core is occupied by recycled material in equilibrium, it is likely that power efficiency would be reduced by about 5%.)

Appendix D: Charter for the AAA Multi-Strata Approaches Study

Objectives

The study will assess the options for a US domestic integrated nuclear waste management strategy based on partitioning (chemical separations) and transmutation, considering the use of available and future nuclear power reactors (fast and thermal spectrum) in combination with fast-spectrum accelerator-driven sub-critical systems. A key step will be development of models representing the flows of materials and waste streams through postulated multi-tier partitioning and transmutation systems.

Background

On April 18, 2001, the Department of Energy agreed to conduct a special analysis requested by the NERAC Subcommittee on the Accelerator Transmutation of Waste. The subcommittee requested that the analysis focus on utilizing a multi-tier (or multi-strata) approach to transmute nuclear waste from commercial reactor spent fuel. The multi-strata approach has potential benefits of reducing the accelerator-driven capacity needed and increasing the transmutation rates by using thermal or fast reactors to accomplish much of the transmutation function.

Methodology

The study will:

- Obtain input on parameters for masses and the form of spent nuclear fuel from the domestic nuclear industry.
- Identify initial approaches based on the literature and existing work by the international and domestic communities.
- Identify critical constraints based on past domestic and international work.
- Create quantitative models, which include actinide and fission product mass balances that can be manipulated against criteria and metrics.
- Develop an initial ranking of the technically feasible approaches against the top-level goals below (or their sub-elements) and estimate technical risk, cost, safety and environmental impact potential. Based on reasonable success criteria, identify two or three preliminary approaches for detailed evaluation.
- Provide initial observations regarding the most attractive approaches to meet top-level and derivative criteria for the waste transmutation mission.

The study will make full use of US industry input into the Gen IV program and solicit their input on the future trends of the domestic industry.

Transmutation Mission-Goals

The following top-level criteria and considerations should be used for ranking approaches. Sub-elements or derivative criteria may be developed for evaluation.

- Improve the long-term public safety by reducing the radiotoxicity and potential radiological dose from spent nuclear fuel.
- Reduce the proliferation risk from the plutonium in spent nuclear fuel.
- Provide benefit to the repository program by reducing the mass and thermal load from spent nuclear fuel.

- Improve the prospects for nuclear power by providing a viable and economically feasible waste management strategy.

Assumptions

Key assumptions for the study will include:

- The work is performed on behalf of the government, and plutonium separation in government facilities for waste management purposes may be considered.
- Plutonium may be burned in commercial reactors for government waste disposal purposes.
- Industry growth rate and mix of generating reactor technologies is based on industry input.
- The study will identify and build on international work—burn-up, loss, and mass flow calculations performed for OECD and Japan.
- The Yucca Mountain repository will operate and store the spent nuclear fuel from existing plants.
- Deployment of technology could begin in about 2010–2015 with the next generation of plants.
- Performances of the various technologies will be assessed on the basis of current knowledge, taking into account reasonable potential improvements from research activities.

Study Implementation

The study will be co-led by the Los Alamos and Argonne National Laboratories. It will be conducted by staff members from those organizations in addition to staff from General Atomics, Burns and Roe, and other laboratories and organizations as appropriate. The study will be guided by a Steering Committee composed of a representative from LANL, ANL, and DOE (Chair).

Resources

Ongoing FY 2001 programmatic activities in the systems area (WBS 1.22) are related to this study and only need to be focused on the activities of the Task Force. Therefore, the efforts of the Task Force are considered in the existing work scope, and there is no need for additional FY 2001 resources.

Schedule

The study will begin in May 2001 and be substantially completed in FY 2001. A report of the results to date and progress will be provided to the NERAC Subcommittee in the fall of 2001.

Reporting

A draft report will be provided to the Steering Committee for review not later than September 15, 2001. The final report, with comment resolution complete, should be issued by November 2001.

_____

John Herczeg, AAA Program Manager
Office of Nuclear Energy, Science, and Technology

Appendix E: Program Goals, Criteria, and Metrics and Potential Impacts on Repository Design and Operation

A primary motivation for the Advanced Accelerator Applications (AAA) Program is the benefit to the geologic disposal of spent nuclear fuel and high-level radioactive waste (HLW). Other motivations include limiting future inventory of plutonium and the potential for sustainable growth of nuclear power through extending fuel resources and minimizing wastes. The top-level programmatic goals include:

- Improving public safety,
- Providing benefits to the repository program,
- Reducing the proliferation risk from plutonium in commercial spent fuel,
- Improving prospects for nuclear power.

Three out of four of these goals relate (at least in part) to beneficial impacts on geologic disposal of waste.

The goals, criteria, and metrics serve to define the AAA Program purpose internally and to communicate the purpose internally. Performance criteria and related metrics may be used in evaluating transmutation concept alternatives and in design studies in combination with other metrics (cost, schedule, efficiency, safety, environment, etc.) To be useful, metrics must be easily applied and simple enough to develop data for, but specific enough to discriminate alternatives.

Improve Public Safety

The most direct impact of transmutation on the repository is the change in the radionuclide inventory to be disposed of, and the replacement of intact spent fuel with one or more engineered waste forms. Removal of most of the uranium and transmutation of most of the actinides normally present in spent nuclear fuel into fission products dramatically alters the radionuclide content and long-term activity of the waste. Converting both the fission products into optimized waste forms offers the potential for longer radionuclide isolation.

The difference in waste streams from transmutation compared to a once-through fuel cycle can be discussed in terms of the reduction in intrinsic hazard levels and the improvement in long-term performance of a geologic repository. The first of these measures, the inherent hazard of the waste measured as radiotoxicity, is independent of the disposal concept. It is comparatively simple to calculate and explain, but does not relate directly to repository safety. The second shows the expected reduction of long-term doses from a conceptual geologic repository. It relates directly to the regulatory requirements for licensing a repository, but requires complex performance assessment calculations.

Radiotoxicity Criterion

Radiotoxicity is frequently used to measure how hazardous a radioactive material is. It represents the potential biological damage to humans if they are exposed to a material in the most damaging way (such as through inhalation or aqueous ingestion). Because different radionuclides give different biological effects, radiotoxicity represents the sum of the radiotoxicities of each isotope present in the material; therefore, radiotoxicity changes as radionuclides decay. It should be noted that radiotoxicity is an inherent property of the material, not considering any barriers to exposure, and thus does not directly reflect public safety or exposure. It also does not directly correspond to repository performance. It is, however, a simple way to compare the inherent hazard of different materials. There are multiple ways of calculating and expressing radiotoxicity, so a consistent method must be used for valid comparison. A typical radiotoxicity includes the worst-case exposure pathway and most critical organ weighting factors to calculate a maximum biological effect per unit of isotopic activity (Sv/Bq).

The objective is to remove and transmute enough of the long-lived radionuclides, primarily actinides, so that after 1,000 years the remaining waste is less radiotoxic than the initial uranium ore (including uranium and the naturally occurring decay products). This criterion is aggressive but feasible because most of the long-term radiotoxicity in spent fuel is in isotopes of plutonium, neptunium, and americium that are readily transmuted.

Repository Dose Criterion

To address the impact of transmutation on the safety of a geologic repository, the performance measure used by the repository program itself must be used. In the US repository program, the regulatory requirements and the licensing safety case are based on estimates of future potential doses to a nearby population. This estimate is radionuclide specific and currently comes from complex performance assessment calculations involving mechanistic and stochastic representations of the various features, events and processes envisioned for the future evolution of the repository following permanent closure. While detailed performance-assessment calculations of transmutation-specific waste streams will eventually be needed, a simpler set of criteria and metrics may be extracted for screening purposes from published performance assessments for the potential Yucca Mountain Repository. The most recent published results are from the *Total System Performance Assessment—Site Recommendation* (TDR-WIS-PA-000001 REV 00 ICN 01A). The base-case dose curves for the most important radionuclides are shown in Figure E-1.

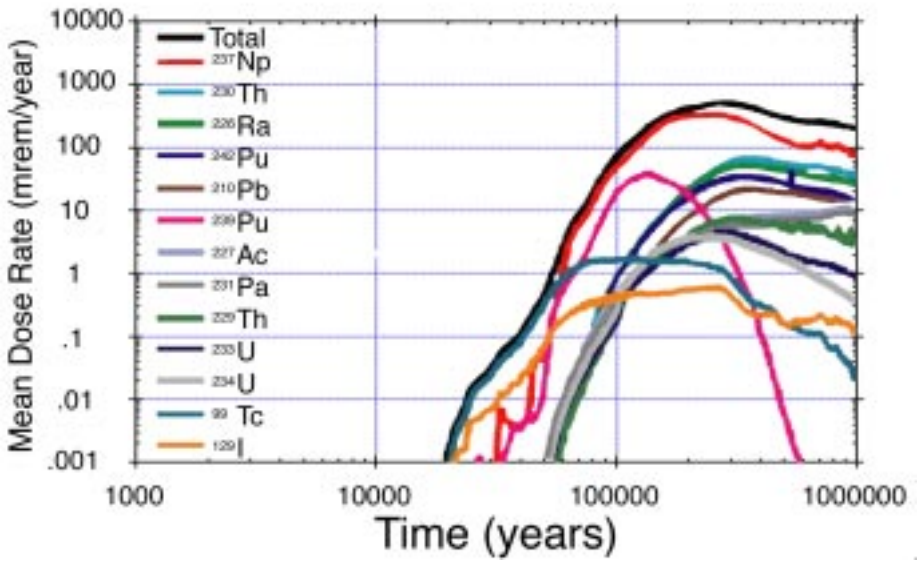


Figure E-1. Calculated 1,000,000-year dose curves for Yucca Mountain TSPA-SR base case. (From "Total System Performance Assessment – Site Recommendation," TDR-WIS-PA-000001 REV 00 ICN 01A, OCRWM M&O, December 2000, page F4-19 Figure 4.1-19a).

The performance-assessment dose calculations are radionuclide specific and account for initial inventory, decay, waste form degradation, aqueous solubility, transportation and retardation processes through various exposure pathways, and biological effect. It is therefore difficult to extrapolate results from one radionuclide to another. It is clear that different radionuclides dominate the dose at different times after repository closure. Tc-99 dominates up to about 50,000 years, after which Np-237 becomes the largest contributor to dose. The onset and shape of the initial dose rise is determined primarily by waste package lifetime and transport times to the accessible environment.

The current regulatory guidance in the proposed EPA 40CFR 197 specifies a dose limit of 150 microsieverts committed effective dose equivalent (CEDE) per year for a period of 10,000 years. While this translates to 15 millirem/year, the CEDE approach is considered approximately equivalent to a 25 millirem/year whole-body dose. The dominant radionuclide in the YMP PA calculations during the first 10,000 years is Tc-99, closely followed by I-129. These two radionuclides are characterized by significant inventory, long half-life, high solubility, rapid release from the engineered barrier system (EBS), and little or no sorption during transport. During the 10,000-year regulatory compliance period, the performance impact of transmutation is directly proportional to changes in the inventory and EBS release rate of these isotopes.

The peak dose, which occurs at about 300,000 years, is dominated by Np-237, followed by Pu-242 about a factor of 5 to 10 lower. These actinides are characterized by significant inventory, high radio-biological effect, long half-life, moderate solubility, slower release from the EBS, and significant retardation during transport. The ability of transmutation to destroy the majority of these actinides would be directly reflected in peak-dose reductions. With transmutable actinides providing more than 99% of the predicted peak dose, this dose-reduction level is an aggressive but viable transmutation criterion.

In terms of repository performance metrics for transmutation, there are several issues that make simple and clear quantitative values difficult to specify. The repository base-case calculation currently meets the proposed regulatory criteria during the 10,000-year regulatory period, caused in large part by slow and distributed failure of the waste packages. In fact, with the current engineered barrier system performance, there may not be any waste-package failures during the 10,000-year regulatory compliance period. If there are failures, the primary concern is Tc-99 inventory and release rate. Once about 60% to 80% of the Tc dose is moderated, the I-129 becomes dominant. The primary effect of transmutation during this period comes from reduced inventory and lower release rate from optimized waste forms, and provides enhanced confidence in the repository safety argument and an alternative to reliance on the engineered waste packages. If Tc and I are not transmuted in the transmutation system, then the initial inventories will increase because of the added fission products formed from fission of the actinides (perhaps a 10%–20% increase). If Tc and I are transmuted, then inventory reduction can be a primary metric. In either approach, containment of these two fission products within a robust engineered waste form could result in a significant performance benefit. This benefit would require waste forms with alteration and leach rates significantly lower than for irradiated UO_2 in spent fuel. If waste-form leach rates are very low, it is possible for this metric to dominate inventory reduction. If the release rate from the EBS is slow compared to 10,000 years, then waste form integrity could dominate repository regulatory compliance.

Reduction in Np inventory is an important criterion because of the opportunity to reduce the peak dose. The proposed regulatory period of 10,000 years does not capture the importance of the Np-237 dose contribution. While the TSPA-VA base-case dose was well below the proposed regulation for much more than 10,000 years, it eventually rises to roughly an order of magnitude above the limit. The importance of this is subject to debate; some people believe that this long-term calculated dose is a severe safety, public perception, and licensing problem, while others dismiss it as beyond the regulatory period. These dose results are also thought to be conservatively bounded, and there is some expectation that refined data and models will reduce the Np-237 dose rate by one or several orders of magnitude. The actinides have another complication; under some repository environmental and water-flow scenarios, their release rate is solubility limited rather than inventory limited. Inventory reduction is, therefore, not reflected in release-rate reduction until the availability drops below the solubility. In that approach, significant inventory reduction might be needed before EBS release rates are reduced. The peak doses from Np typically do not suffer from this solubility limit. Despite these complexities, significant reduction in the Np inventory would be a major benefit to the repository safety argument. It is beneficial to reduce the largest contributor to population dose, particularly a dose that rises well above the proposed regulatory limit, albeit after the regulatory time period. Inventory reduction provides the primary benefit because the time

scale for this peak dose is extremely long, and demonstrating a waste form with a low release rate for geologic times would be difficult.

In the event that orders of magnitude benefit are gained in the metrics discussed above, it is useful to see what other radionuclides and waste streams might become important. Insight into contributions from different waste streams can be found in the performance assessment-supporting document for the ATW Roadmap[2]. In the first 8,000 years, ATW reduction in the base-case commercial spent fuel (CSNF) fission-product dose is only limited by the C-14 dose, which is nearly two orders of magnitude below Tc-99. Toward the end of the proposed 10,000-year regulatory compliance period the dose contribution from DOE-owned spent fuel (DSNF) rises to within a factor of five of the CSNF dose. This implies if only CSNF is processed through transmutation, fission-product dose reduction greater than 80% becomes less important after about 8,000 – 10,000 years. Transmutation processing of DSNF provides further dose reduction until after 12,000 years at which time dose contributions from vitrified high-level radioactive waste (HLW) rise to about one-tenth the dose from CSNF. This timing is beyond the 10,000-year regulatory period, but indicates that more than a 90% reduction in CSNF fission products and 50% reduction in DNSF fission products become less important. After 50,000 years, where Np-237 dominates the dose, a similar situation exists. Transmutation reduction of the CSNF actinides runs into diminishing returns from DSNF after a one-and-one-half order of magnitude reduction, with HLW further limiting dose reduction to about one order of magnitude in the 50,000-year to 500,000-year period. Below DSNF and HLW limits to dose come from uranium-234, which transmutation processing would divert from the repository, and from immobilized disposition of excess weapons-grade plutonium. These later effects are probably unimportant, as transmutation would have to have a three-to-four orders of magnitude reduction in SNF and DSNF actinides and fission-product doses to make other contributions significant.

Provide Benefits to the Repository Program

Processing and transmutation of spent nuclear fuel provides the repository program with potential benefits in a variety of design and licensing areas. These include flexibility in selection of waste form, waste-package design, management of decay heat and criticality control. Separation and separate management of the uranium as either a resource or non-high-level waste offers the opportunity for reduced mass and/or volume of repository waste. While most of these are difficult to quantify, they represent significant benefits and warrant continued evaluation.

Heat-Load Criterion

An important design issue for the repository that would be modified by transmutation processing is the thermal load from decay heat. The waste-package design includes internal structures for heat transfer, and there are design constraints on temperatures for the cladding, waste-package surface, host rock wall, and all the way to the ground surface above the repository. For the inventory envisioned for the US geologic repository, fission-product decay (primarily Cs-137 and Sr-90, with half-lives of about 30 years) dominate the thermal output for about the first 80 years. Waste-package surface temperatures can exceed boiling temperatures unless adequate cooling is provided. At longer times, actinide decay dominates the thermal output, progressing through a series of actinides from a shorter to longer half-life. While the long-term thermal output is low compared to the early time, once the repository is closed the heat is retained until temperature gradients are sufficient to conduct heat through hundreds of meters of rock. In general terms, fission-product decay heat is a design and handling issue during the operational period of a repository, and actinide decay heat is a design and performance issue for geologic times following repository closure. Some nations anticipate extended cooling of either spent nuclear fuel or processed HLW to allow cooling, but current US policy is for prompt waste disposal rather than storage.

As discussed in chapter 8 of the “Systems, Scenarios, and Technical Integration Working Group Report for the ATW Roadmap,” transmutation can have a major impact on the thermal response of a repository. With transmutation, most of the actinides are fissioned, resulting in more fission products but very little actinide inventory. This would skew the thermal output curve strongly to the early years; however, the

fission products spend a significant residence time in the transmuter systems, thus allowing for more decay. With removal of the long-lived actinide heat, and viable options for short-term cooling, transmutation provides a great deal of flexibility in thermal management that does not exist for direct disposal of spent fuel with the actinide inventory intact. This flexibility has potential benefits for repository design, waste package design, underground layout, ventilation, repository operations, and long-term performance. With destruction of most of the actinides, an order-of-magnitude reduction in the long-term heat load is feasible.

Criticality Criterion

The potential for a criticality event in or near a geologic spent fuel repository has been the subject of considerable debate and analysis over the past decade. While the circumstances that could lead to criticality generally seem to be of quite low probability and limited consequences, the issue continues to be raised. In response, the US repository program continues to invest resources in criticality and consequence analysis. Criticality control also places constraints on waste package design and includes engineered criticality control materials. If transmutation removes most of the fissile material from the repository inventory, criticality should rapidly become a non-issue. With criticality scenarios already requiring unlikely conditions and processes to approach feasibility, reduction of the fissile inventory by one or more orders of magnitude would make them implausible or even impossible. The benefits of eliminating criticality concerns would primarily be in public perception and licensing confidence. An additional benefit would occur in eliminating design and material performance requirements. Criticality concerns include highly enriched uranium and excess weapons fissile materials in the repository, as well as low-enriched uranium and plutonium in commercial fuel.

Mass Criterion

Processing spent nuclear fuel and removing the uranium immediately changes repository design constraints on mass, volume and shape of the waste. The waste package is no longer required to accommodate 5-meter-long fuel assemblies, so other design aspects can optimize waste package size, shape, and content. Removal of the uranium to another disposition path eliminates more than 90% of the mass and much of the volume. The final mass, volume, physical form, and thermal and radiation properties of the waste steams are yet to be determined, but a great deal of flexibility exists to accommodate repository optimization. Total waste mass will be determined by the immobilization matrix and radionuclide concentration selected for each waste stream. The long-term isolation benefit from optimized waste forms allows performance benefits. The waste streams from transmutation could also be more homogeneous than the current range of CSNF, DSNF, and HLW that constrain the current repository design.

Repository cost and capacity considerations are important but complicated questions. Costs will depend on potential changes in repository design, waste-form and waste-package design, thermal management, policy and regulatory evolution, and repository capacity. The current US waste-management program evolved to manage a specific set of SNF and HLW. A future fuel cycle, including transmutation, would probably result in further evolution of that program to accommodate the technical impacts from different wastes. Initial repository capacity is a legislated value of 70,000 metric tons taken as initial heavy metal equivalent; thus, removal of either the uranium or the actinides does not change the specified capacity. There would be less waste, less hazard, optimized forms, and less long-term thermal load might, but still the same initial heavy metal equivalent. Nuclear fuel cycles using transmutation would provide a revised technical basis in the event there were future evolution of the legal, policy and regulatory framework for geologic disposal, however.

Reduce the Proliferation Risk from Plutonium in Commercial Reactor Spent Fuel

Commercial nuclear power currently produces a steadily increasing global inventory of plutonium. Much of this plutonium is dispersed in spent fuel awaiting disposal or other disposition, and some has been

separated and is stored for future use as fuel. Some is being consumed in MOX fuel, but the destruction rate is small. Any transmutation-based fuel cycle would offer the opportunity to slow, halt, or even reverse this accumulation.

Plutonium Inventory Criterion

Continued use of the once-through fuel cycle will result in an increasing quantity of plutonium requiring disposal and safeguards. A fundamental transmutation criterion is to consume plutonium at a rate significant enough to reverse this accumulation.

Plutonium Disposal Criterion

Proliferation risk from geologic disposal of spent fuel is subject to debate and wide differences in opinion. The proposed repository inventory for Yucca Mountain includes several hundred tons of reactor-grade plutonium distributed over 63,000 tons of CSNF. The desirability, as well as the accessibility, of this material is both important and controversial. While it is acknowledged that reactor grade Pu might be usable to produce a nuclear explosion, it is considered less desirable for military systems than weapon-grade plutonium. Accessibility is limited by the deep geologic setting, heat, and radiation generated by spent fuel and by the fuel form itself; however, the isotopic desirability and accessibility both improve with time, resulting in a potentially more attractive material in the distant future. This has resulted in some characterization of a spent fuel repository as the *plutonium mine of the future*. In addition to Pu in CSNF, other potentially attractive nuclear materials are scheduled for geologic disposal, including high-enrichment uranium fuels, immobilized excess weapon-grade plutonium and significant quantities of other fissile actinides.

Transmutation has great leverage over the fissile materials because of their large fission cross sections. Achieving at least an order of magnitude reduction in fissile material inventories nearly eliminates arguments of proliferation risk at the repository. Not only is the attractive material severely reduced in quantity, but it is also mixed with a larger fraction of nonfissile isotopes. Since the repository proliferation risk itself is not well quantified, defining the proliferation benefit is difficult. A two-order-of-magnitude reduction in plutonium and a 75% reduction in the fissile content of the remaining plutonium appear to be aggressive but viable criteria, however.

Plutonium Accessibility Criterion

In the processes required for transmutation, plutonium and other fissile materials must be handled. This could result in an increased risk of diversion unless inherently diversion-resistant processes and adequate safeguards are applied. An objective of the transmutation program is to minimize the risk of diversion and to apply effective safeguards.

Improve Prospects for Nuclear Power

While geologic repositories for the direct disposal of spent fuel are progressing in several countries, including the US, the AAA program provides a potential alternative. This alternative provides both a back up if needed, as well as the opportunity for a fuel cycle that uses resources more efficiently and produces less waste and waste of lower long-term hazards. This becomes increasingly important and cost effective if the use of nuclear energy increases and continues. Long-term nuclear-energy sustainability can be strongly enhanced with a transmutation-based fuel cycle.

Viability Criterion

To be cost effective, a transmutation-based fuel cycle should maximize the generation of electricity in reliable reactor systems while extracting the maximum energy from mined uranium and minimizing the waste-disposal requirement.

Appendix F: Assumptions Regarding the Reactors/Transmuters Evaluated

In this study, a separate commercial nuclear power production sector is assumed, which provides a sustained feed of spent nuclear fuel; initial results focus on materials derived from an evolutionary advanced light-water reactor (ALWR). This spent fuel is treated in a subsequent government-owned waste transmutation enterprise to reduce the long-lived toxicity, reduce the repository radiation dose, and reduce the quantity (and attractiveness) of the fissile material. As discussed in Section 3, the transmutation system may be composed of multiple tiers. For the scenarios investigated in this study, a first-tier thermal reactor system was proposed; both ALWR and advanced gas cooled reactor (GT-MHR) technologies were considered. In all approaches, a final tier fast-spectrum system was utilized; both accelerator-driven and fast reactor options were considered. In this appendix, the principal characteristics are described for the five distinct system types noted above: commercial ALWR, transmutation ALWR, GT-MHR, fast-spectrum ADS (i.e., ATW), and advanced fast reactor (ALMR).

Commercial Advanced Light-Water Reactor (ALWR)

For the commercial system, it is presumed that an evolutionary LWR design will be employed. The precise design parameters of these power production systems will be determined by commercial concerns. The spent fuel discharged from this system will provide the feed material for the transmutation mission. A key assumption is that enriched uranium fuel will be utilized, but a discharge burnup level higher than current LWRs will be achieved. A burnup of 50,000 MWd per metric ton as targeted by modern fuel development programs was assumed.

The discharge fuel composition was based on a previous high burnup PWR spent fuel evaluation. In particular, the extended burnup PWR model developed in the Yucca Mountain spent fuel evaluations (documented in ORNL/TM-11018) was used. This model is based on detailed depletion computations for a *generic* PWR assembly design using a higher enrichment (4.2% U-235/U) to obtain the higher burnup level.

Transmutation Advanced Light-Water Reactor (ALWR)

For the transmutation system, it is presumed that an ALWR design is adapted for the transmutation mission to use 100% core loading of alternate (transmutation) fuel forms. In particular, mixed oxide (MOX) fuel and nonfertile fuel (NFF) forms are considered (see Appendix J for more detail). The system design is based on the 100% MOX and NFF full-core problems that were analyzed by the Paul Scherrer Institute in *Nuclear Technology* (1998). The ALWR lattice calculations employed in this study (see Appendix K) were benchmarked against the PSI core results.

Lattice and reactor specifications used in the ALWR-tier analyses are provided in Table F-1. The *standard* lattice (pin pitch = 1.430 cm) has a moderator-to-fuel volume ratio of 2.0, which is typical for an LWR lattice loaded with uranium-oxide (UOX) fuel. The primary differences between the MOX and NFF core specifications are the fuel form, the fuel-assembly loading, and the fuel-pin geometry. The nonfertile fuel design employs an annular fuel design that has a ZrO_2 zone at the center of the fuel pin. The presence of the diluent and differences in fuel form result in a much lower heavy metal loading for the NFF assembly (as shown in Table F-1). A *highly moderated* lattice, with a moderator-to-fuel ratio of 3.5 (pin pitch = 1.726 cm), was also considered in this study. The increased neutron moderation in this lattice increases the soluble boron's worth and provides a slight improvement to the void coefficient. The standard lattice was used for the final set of ALWR Tier 1 results, however.

Table F-1. Lattice and Reactor Specifications for ALWR Analyses

Parameter	MOX	NFF
Number of pins/assembly	225	
Pin pitch (cm)		
Standard lattice	1.430	
Highly-moderated lattice	1.726	
Inter-assembly gap (cm)	0.11	
Number of fuel pins/assembly	205	
Fuel pin I.D. (cm)	n.a.	0.4565
Fuel pin O.D. (cm)	0.930	
Fuel pin clad O.D. (cm)	1.075	
Active fuel length (cm)	359.5	
Number of water holes/assembly	20	
Water hole I.D. (cm)	1.16	
Water hole O.D. (cm)	1.24	
Assembly HM loading (t)	0.4422	0.0436
Number of assemblies in reactor	177	
Reactor power (MWt)	3000	

Transmutation Advanced Gas Reactor (GT-MHR)

For the transmutation system, it is presumed that a General Atomics GT-MHR design will be adapted for the transmutation mission to use 100% core loading of particles based on the *conventional* TRISO design, but with a transuranic or plutonium-oxide fuel kernel. The key parameters for the fuel particles are summarized in Table F-2.

Table F-2. Parameters for GT-MHR TRISO-Coated Particle Fuel

Region	Composition	Radius (μm)
Fuel kernel	(TRU or Pu) $O_{1.7}$	150
Buffer	Low-density graphite	250
Inner pyro-carbon layer	High-density graphite	285
Silicon carbide	Sic	320
Outer pyro-carbon layer	High-density graphite	360

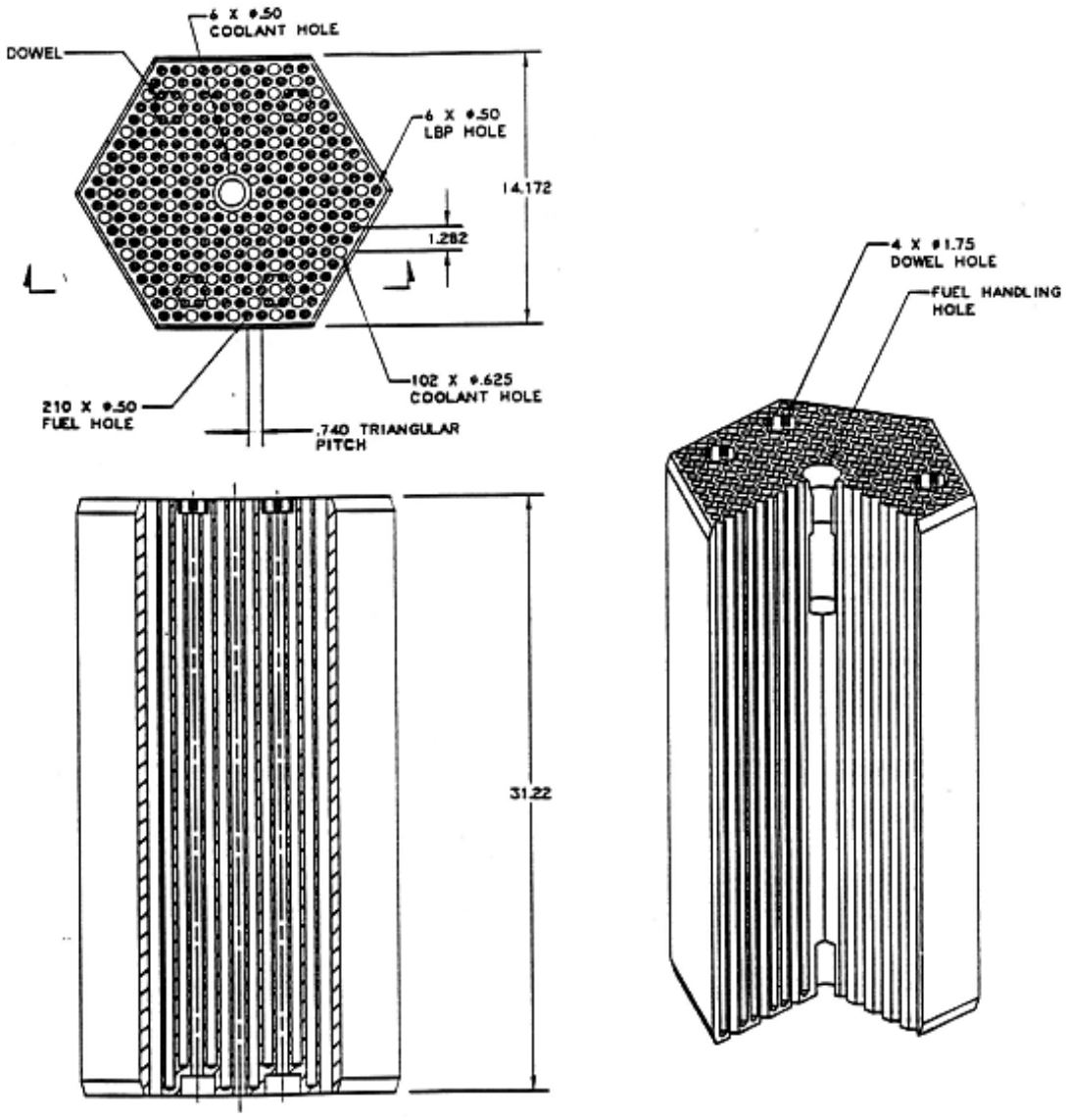


Figure F-1. GT-MHR fuel block.

The reactor design is identical to the standard 600 MWt GT-MHR, as shown in Figures F-1 and F-2. Each fuel element block has provision for 216 fuel/burnable poison channels and 102 coolant passages (cf. Figure F-1). Ten blocks are stacked to form a fuel column, and 102 fuel columns are formed in three rings between inner and outer graphite reflectors to form the active core, as shown in Figure F-2.

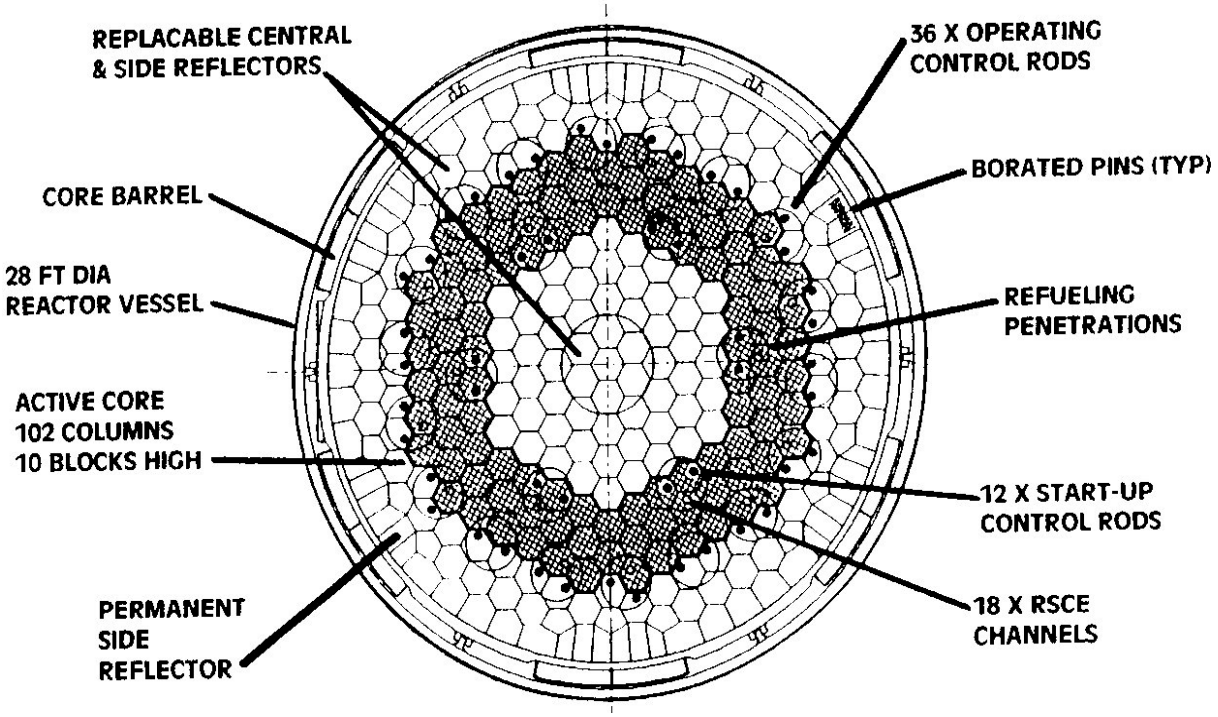


Figure F-2. Configuration of the GT-MHR 600 MWt reactor core.

Accelerator-Driven Fast-Spectrum System (ATW)

For the ADS system, the ATW system-point design employing a sodium-cooled transmutation blanket was used; the development of this design is documented in detail in the AAA system-point design compendium technical report, AAA-RPO-SYS-01-0008. We assume a non-uranium metallic alloy (TRU with ~40% Zr) will fuel the subcritical transmutation blanket; pyrochemical techniques are used to recycle residual transuranics in this fuel after irradiation. A transmuter fission power level of 840 MWt was assumed; this is the same power level previously adopted for the PRISM Advanced Liquid Metal Reactor (ALMR). Parametric studies were performed to optimize the sizing of the sodium-cooled transmuter blanket, to mitigate power peaking problems near the source region, and to assess startup core performance.

The point-design layout, shown in Figure F-3, consists of 19 hexagonal lattice positions containing the lead-bismuth eutectic target/buffer and 132 fuel assemblies, and is surrounded by two hexagonal rows of steel reflector assemblies and one row of B4C shield assemblies. Dividing the blanket into two *enrichment* zones and optimizing the relative TRU loading in each zone will flatten the blanket power distribution. The principal design parameters of the sodium-cooled ATW configuration are summarized in Table F-3. For this study, the overall system size of the point design was maintained; the active height and assembly pitch remain constant. The variations in feed enrichment require changes in the fuel volume fraction, however, to achieve the required subcritical levels and metal alloy enrichments (see Appendix K for details).

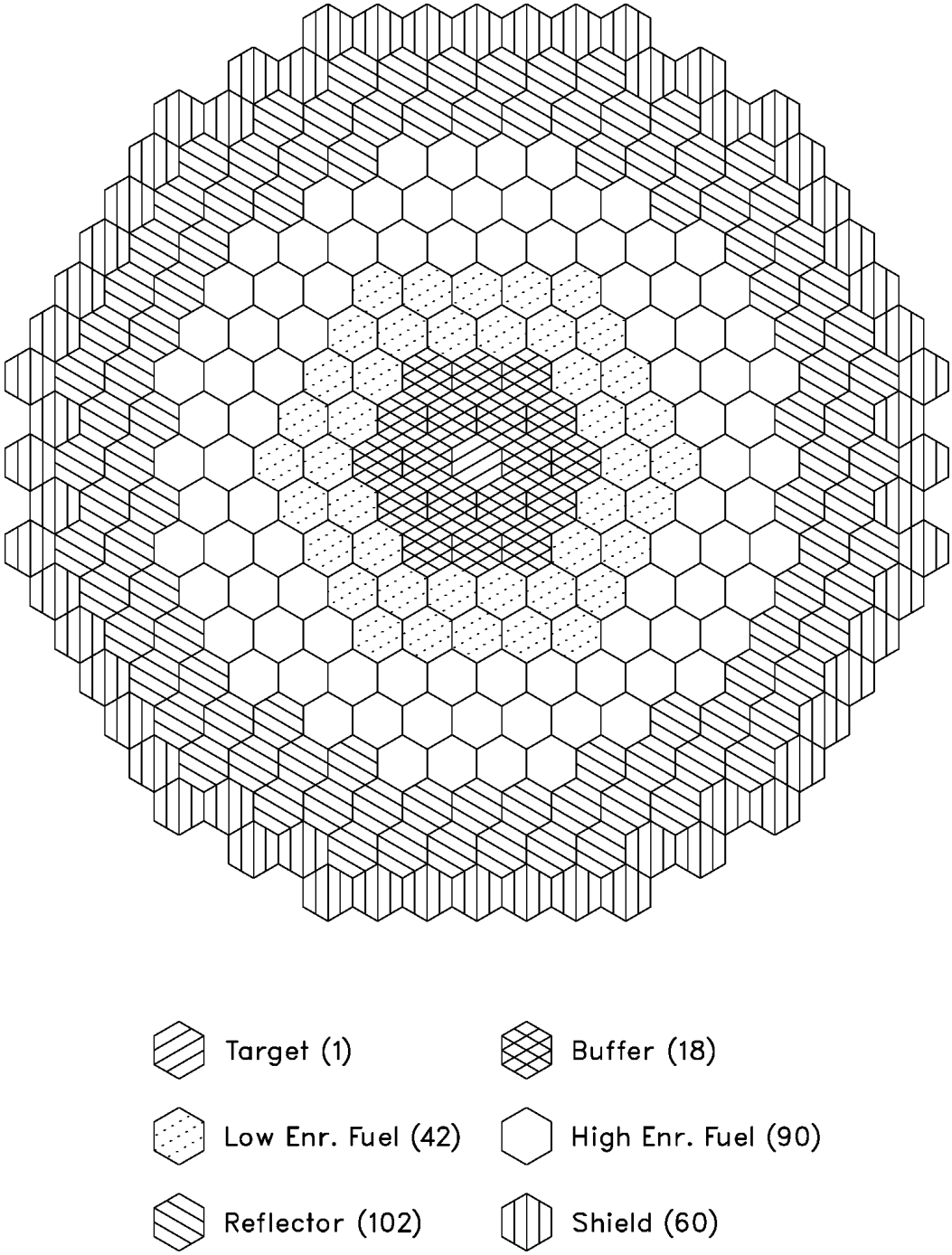


Figure F-3. ATW sodium-cooled system-point design configuration.

Table F-3. Design Parameters of ATW Sodium-Cooled Blanket System-Point Design

Proton energy (GeV)			1.0
Target material			LBE
Fuel material			TRU-40Zr
Pin diameter (cm)			TBD
Number of pins per assembly			271
Fuel smear density (%)			85
Hexagonal assembly pitch (cm)			16.14
Number of assemblies	LBE target/buffer		19
	Fuel	Inner zone	42
		Outer zone	90
		Total	132
	Reflector		102
	Shield		60
TRU fraction split factor (outer zone/inner zone)			1.3
Active fuel height (cm)			107
Equivalent fuel region diameter (cm)			208
Maximum blanket diameter (cm)			300
Number of fuel batches	Inner zone		7
	Outer zone		8
Cycle irradiation time (day)			135

This variation in fuel-volume fraction was achieved by varying the fuel-pin size, effectively exchanging fuel in the close-packed lattice for more coolant. The point-design fuel management strategy is also retained. An eight-batch scheme with semiannual refueling, staggered reloading of neighboring assemblies, and no fuel shuffling is employed for the outer zone; the fuel residence time in the inner blanket zone is reduced to seven cycles to limit the fluence.

Transmutation Fast Reactor (ALMR)

For the transmutation fast reactor, the 840 MWt Advanced Liquid Metal Reactor (ALMR) design was used for this study. This design was developed in the former US fast reactor program by General Electric and Argonne National Laboratory in the 1985 – 1995 time frame; conventional and burner configurations for the weapons plutonium disposition mission are described in detail in Hill (1995). For this study the conventional burner configuration was adapted to the transmutation mission, as described below. Note that the ALMR fuel assembly design was the source for the assembly design used in the ATW system-point design. For the ALMR, the original close-packed fuel lattice with 0.293 inch pins and a 1.197 pitch-to-diameter ratio was retained; thus, the assembly dimensions are identical to the ATW design in this study, but the ALMR fuel volume fraction is higher.

The ALMR conventional burner design uses standard fertile-fuel forms—ternary metal U/TRU-10Zr alloy with maximum TRU content of ~30%. The core configuration is shown in Figure F-4. Reducing the core height to 18 inches *spoiled* the core geometry in this configuration. This increases the leakage and

reduces the conversion ratio (CR) to ~0.5, allowing net consumption of the TRU feed at roughly half the rate of a pure burner (i.e., ATW) system.

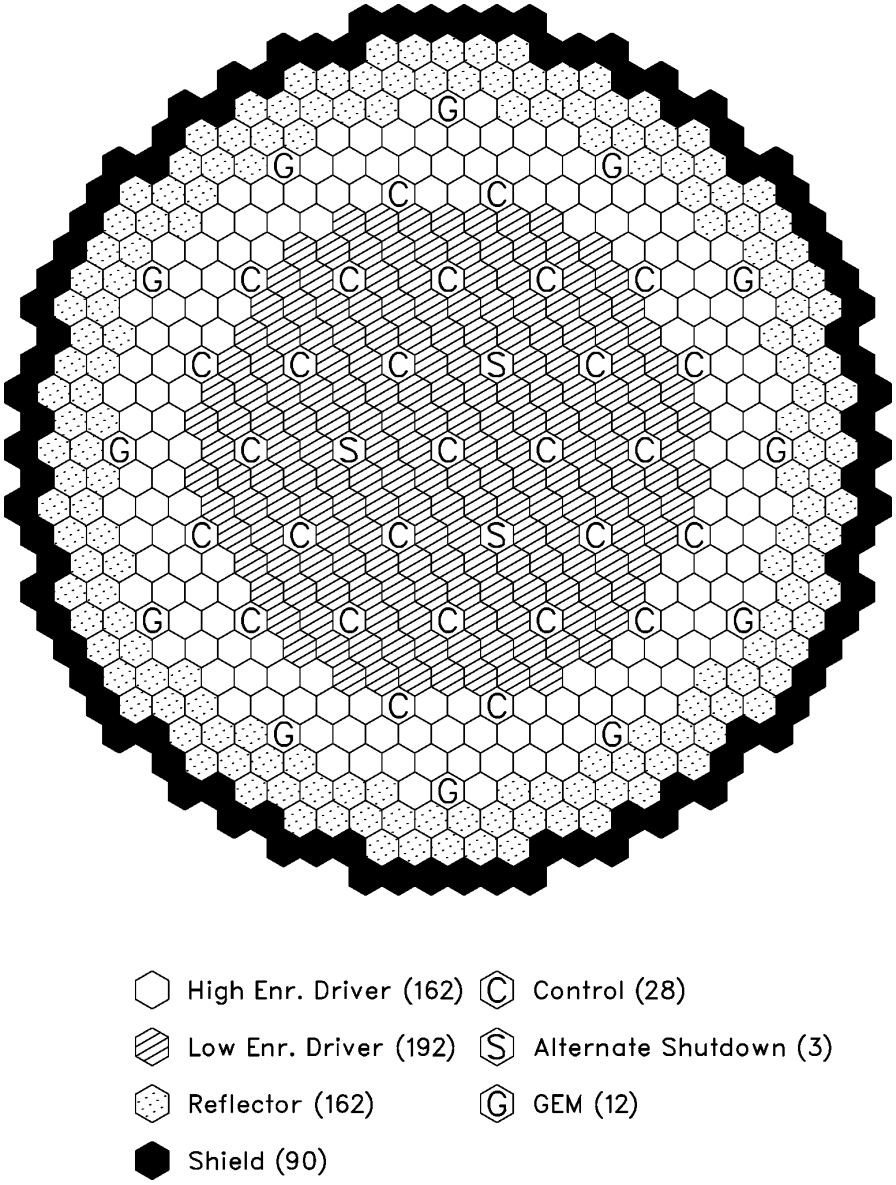


Figure F-4. ALMR burner core configuration.

Appendix G: Flow Charts for Nine Approaches Analyzed in This Study

Three fundamentally different approaches to waste transmutation, described by number of tiers and plutonium separation status (from minor actinides), were addressed in these analyses. The numbering scheme is shown below in Table G-1.

Table G-1. Main Approach Numbering Scheme

	Pu Separated from MA	Pu and MA Remain Together
Two-tier Variations	1 1X, 1Z, 1G, 1XT	2 2X, 2Z, 2G
Single-Tier Variations	N/A	3 3M, 3T

For the two-tier approaches (1 and 2), variations are labeled according to their fuel form. Those employing fertile mixed-oxide (MOX) fuel are denoted with an "X." Oxides embedded in an inert ZrO_2 matrix are denoted with a "Z." Options employing TRISO particles are denoted with a "G," as these are used in gas reactors. The 1XT approach differs from the other Approach 1 situations in that the second-tier is assumed to be a fast reactor rather than the accelerator-driven system employed in all other two-tier approaches. The single-tier approach (3) includes two fast-spectrum variations; the accelerator-driven alternative is denoted with an "M," while the fast-reactor approach is denoted with a "T." These approaches are shown in relation to each other on a tree-diagram below (Figure G-1), and are described individually on the pages that follow.

The primary purpose of the approach-specific tables in this section is to aid the reader in distinguishing among the various approaches to waste transmutation addressed within this study.

Approach Tree

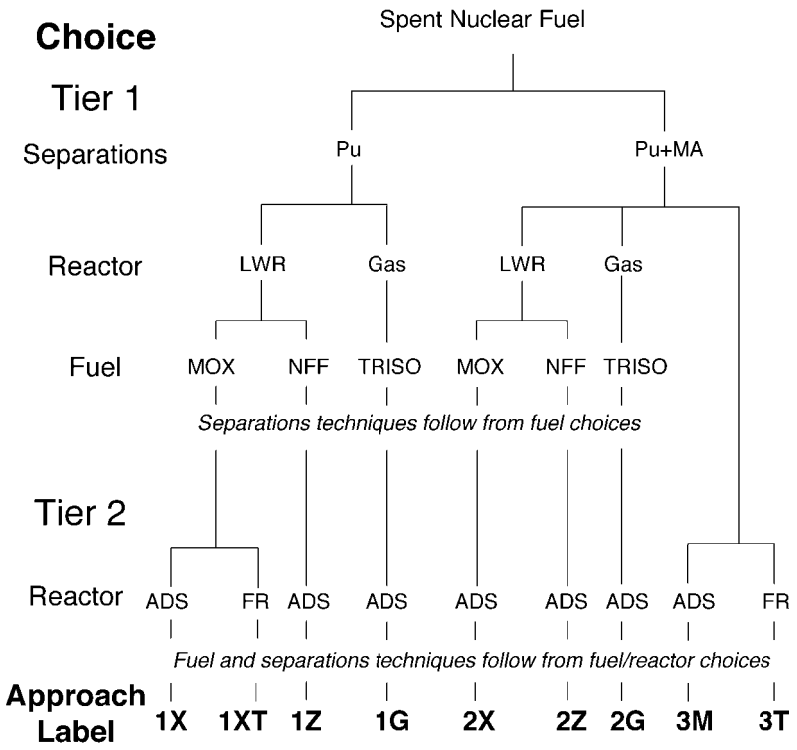


Figure G-1. Labeling scheme for the nine approaches considered.

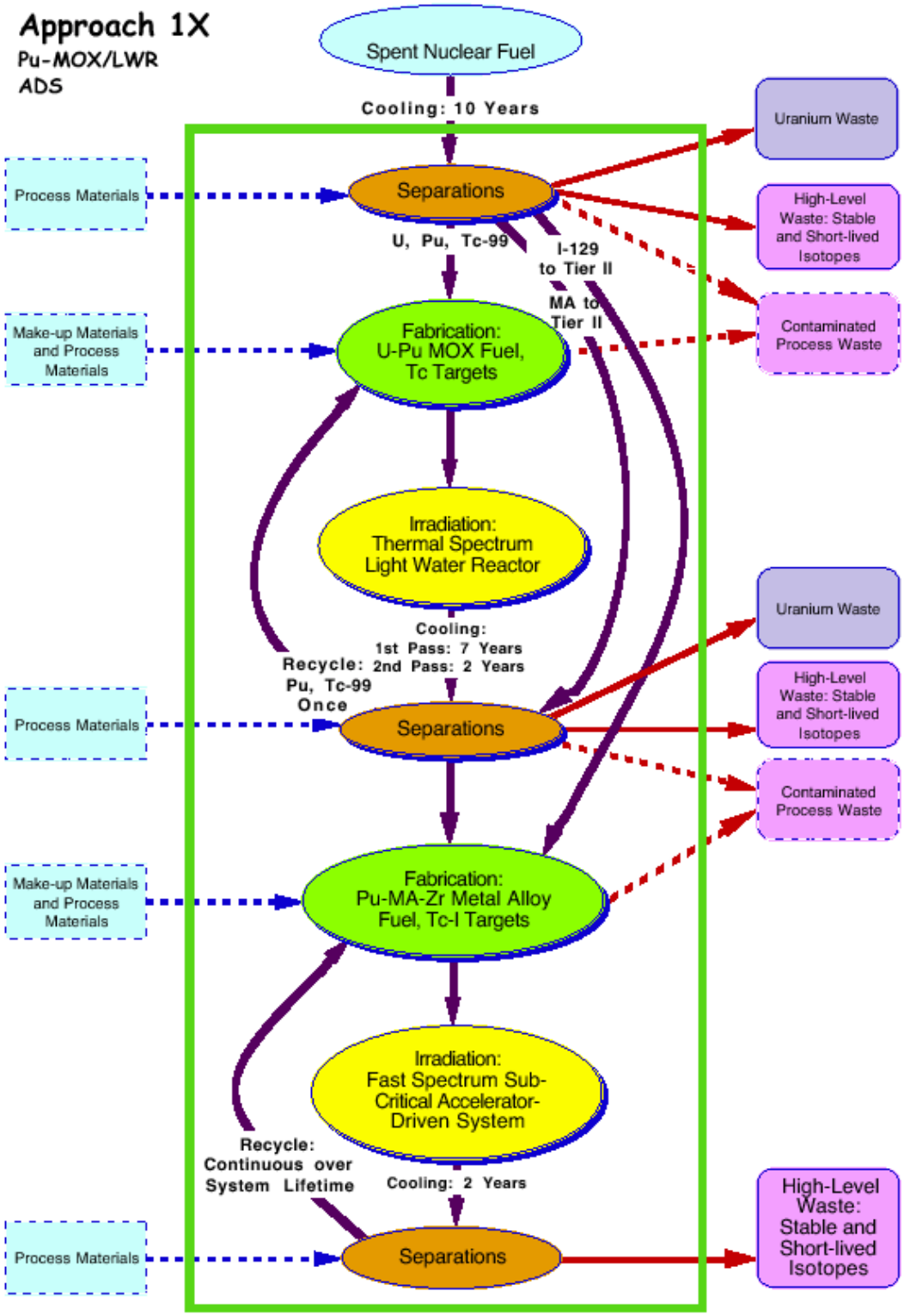


Figure G-2. Flow chart for Approach 1X, which employs a uranium-plutonium mixed-oxide fuel and assumes one recycle at Tier 1.

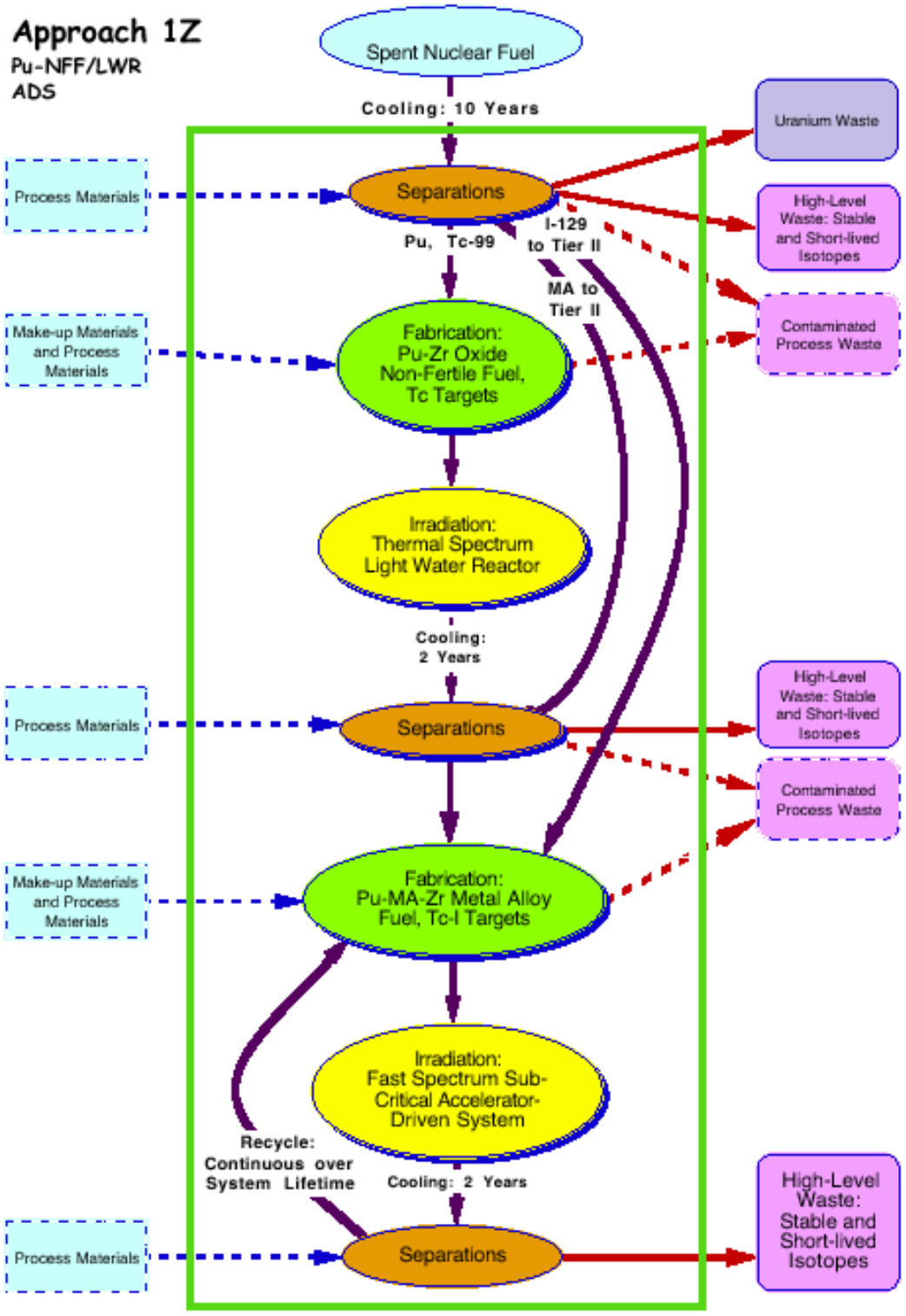


Figure G-3. Flow chart for Approach 1Z.The primary difference between 1X and 1Z is the Tier 1 fuel choice.

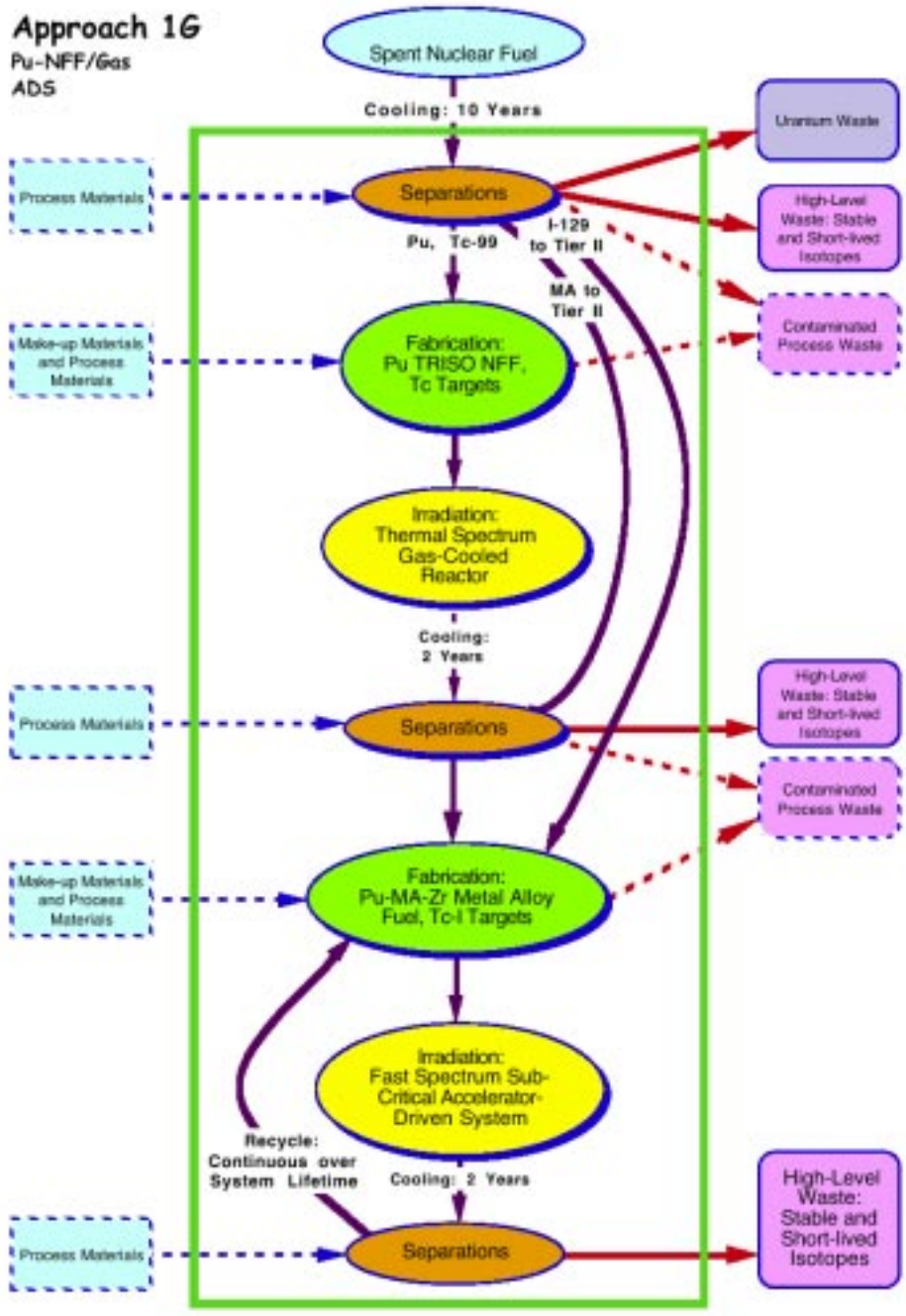


Figure G-4. Flow chart for Approach 1G, which introduces a different Tier 1 fuel choice: TRISO-coated nonfertile fuels. Additionally, these fuels are burned in a thermal-spectrum gas-cooled reactor rather than a light-water reactor, as was the situation for Approaches 1X and 1Z. Potential inclusion of Np with Pu would reduce the amount of material directly charged to Tier 2.

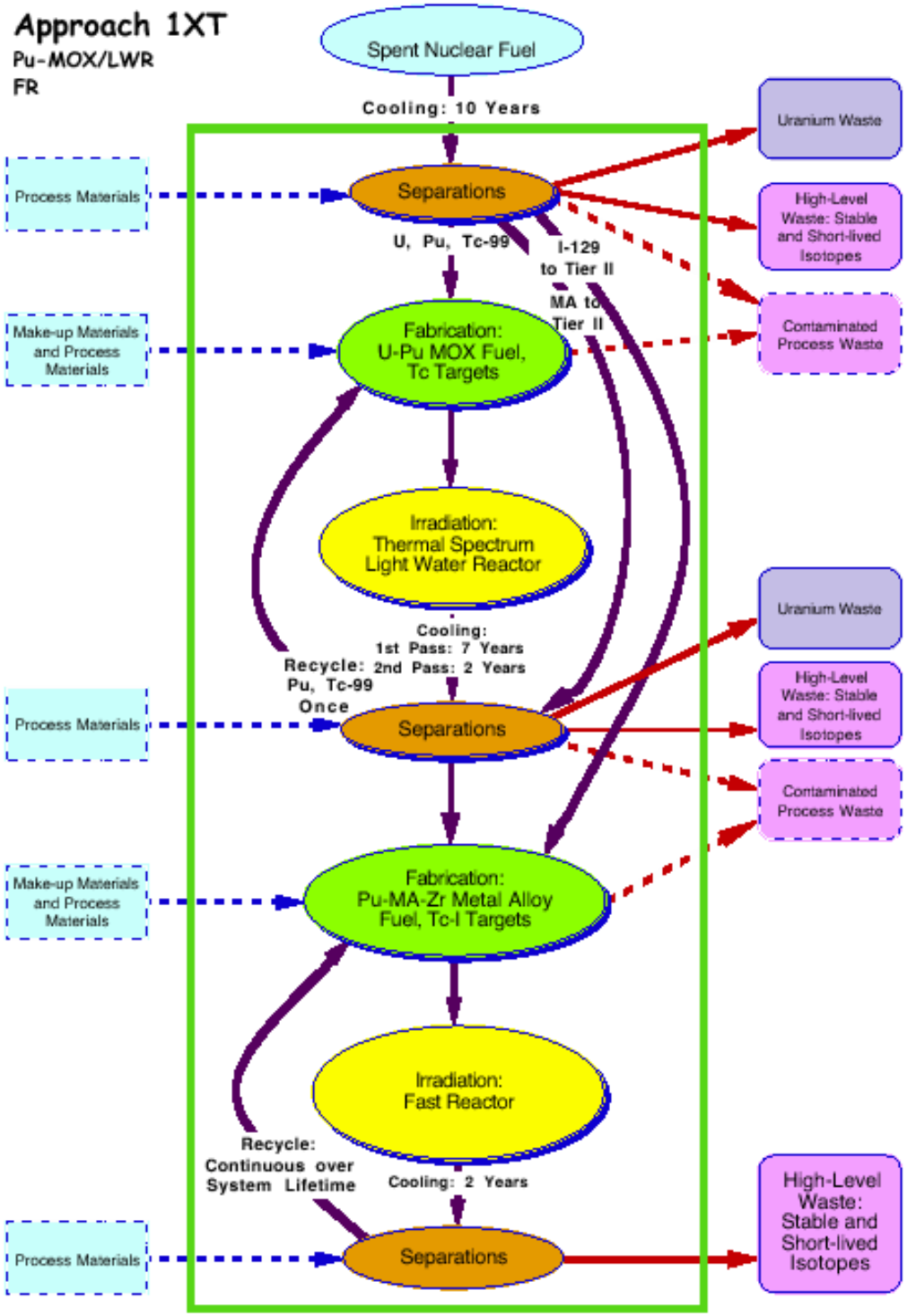


Figure G-5. Flow chart for Approach 1XT, a modification of Approach 1X, in which the second tier is a fast reactor rather than an accelerator-driven system. Like Approach 1X, it employs a uranium-plutonium mixed-oxide fuel and assumes one recycle at Tier 1.

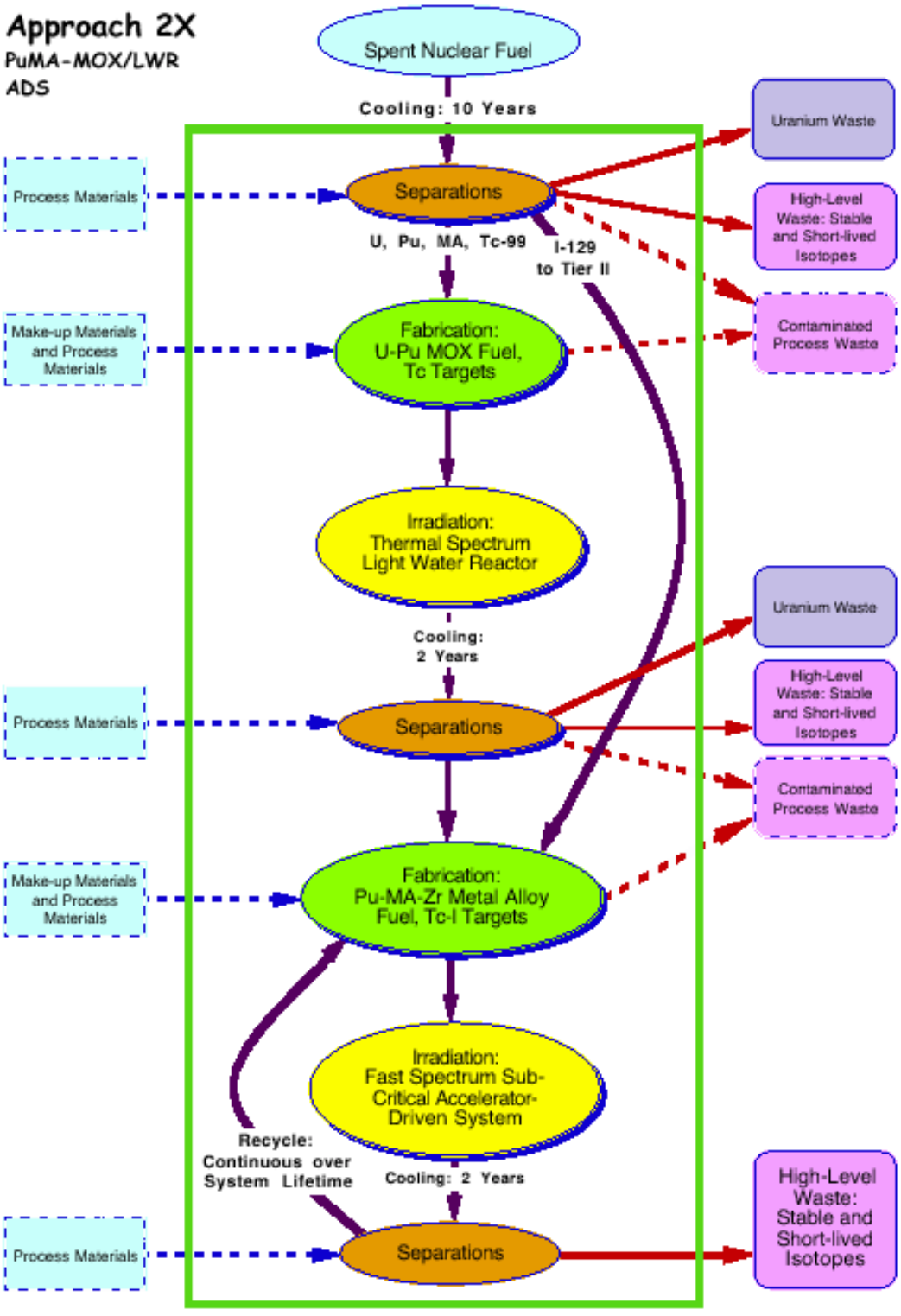


Figure G-6. Flow chart for Approach 2X, which differs from Approach 1X in that the minor actinides in the commercial spent fuel are not separated from the plutonium.

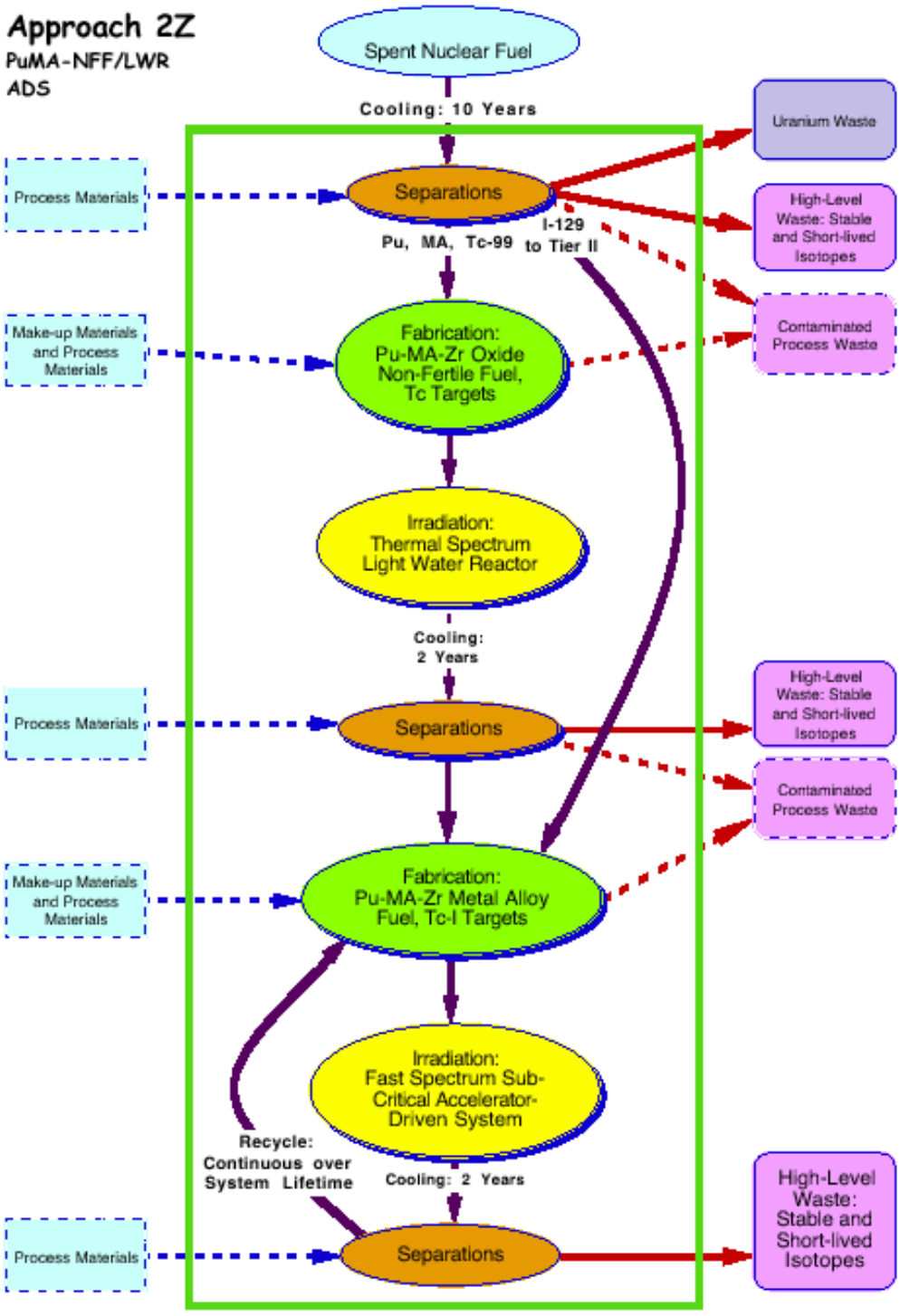


Figure G-7. Flow chart for Approach 2Z. This fuel form is a Pu-MA-Zr oxide (nonfertile) rather than U-Pu-MA mixed oxide. As with Approach 2X, the minor actinides in the commercial spent fuel are not separated from the plutonium.

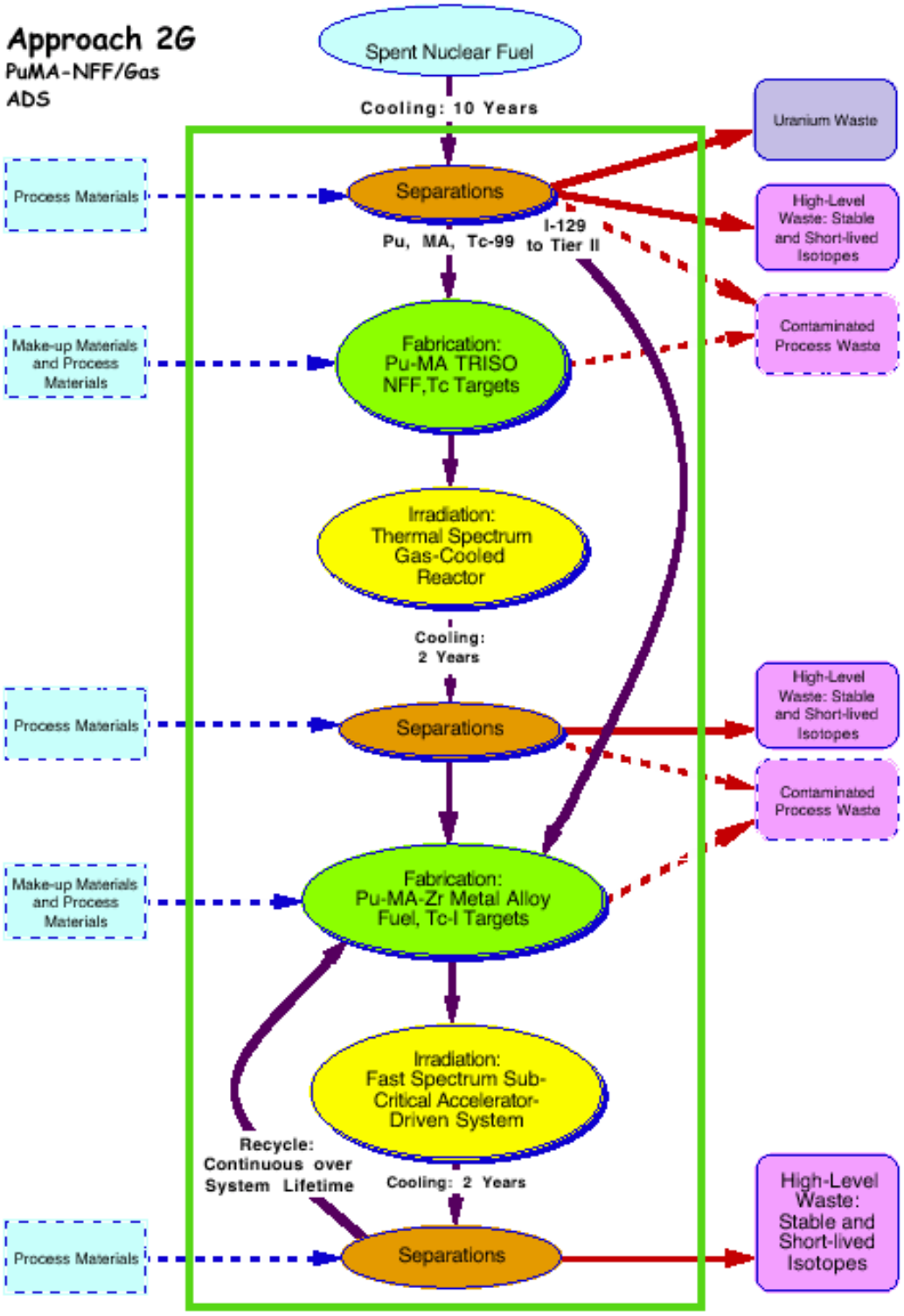


Figure G-8. Flow chart for Approach 2G. As with its nonseparated counterpart (Approach 1G), Approach 2G employs TRISO-coated nonfertile fuels. Additionally, these fuels are burned in a thermal-spectrum gas-cooled reactor rather than a light-water reactor.

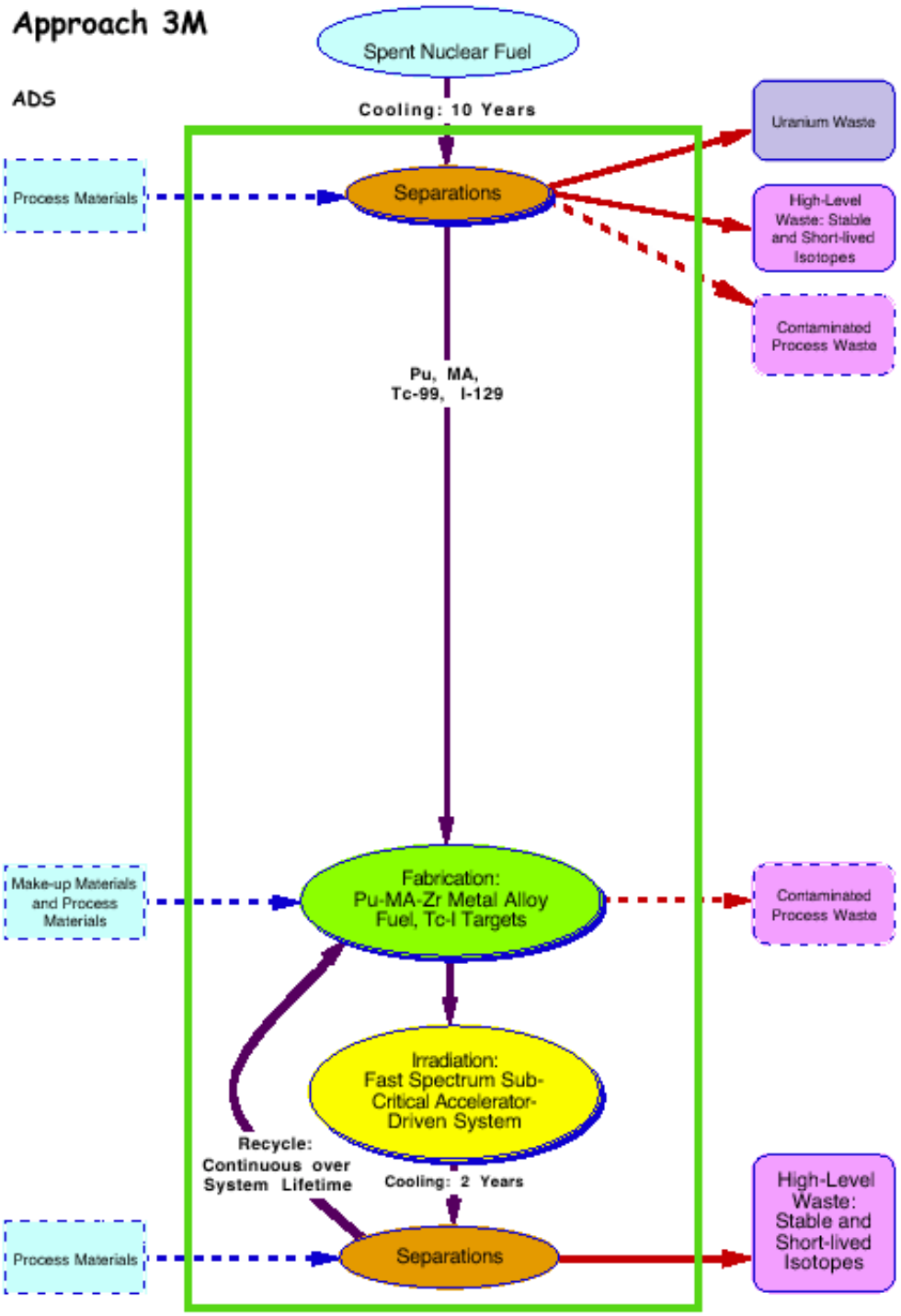


Figure G-9. Flow chart for Approach 3M, which differs from the preceding approaches in that there is no first (thermal-spectrum) tier. Plutonium and minor actinides in the separated commercial fuel go immediately to a fast-spectrum accelerator-driven system. This is quite similar to the 1999 ATW Road Map reference case.

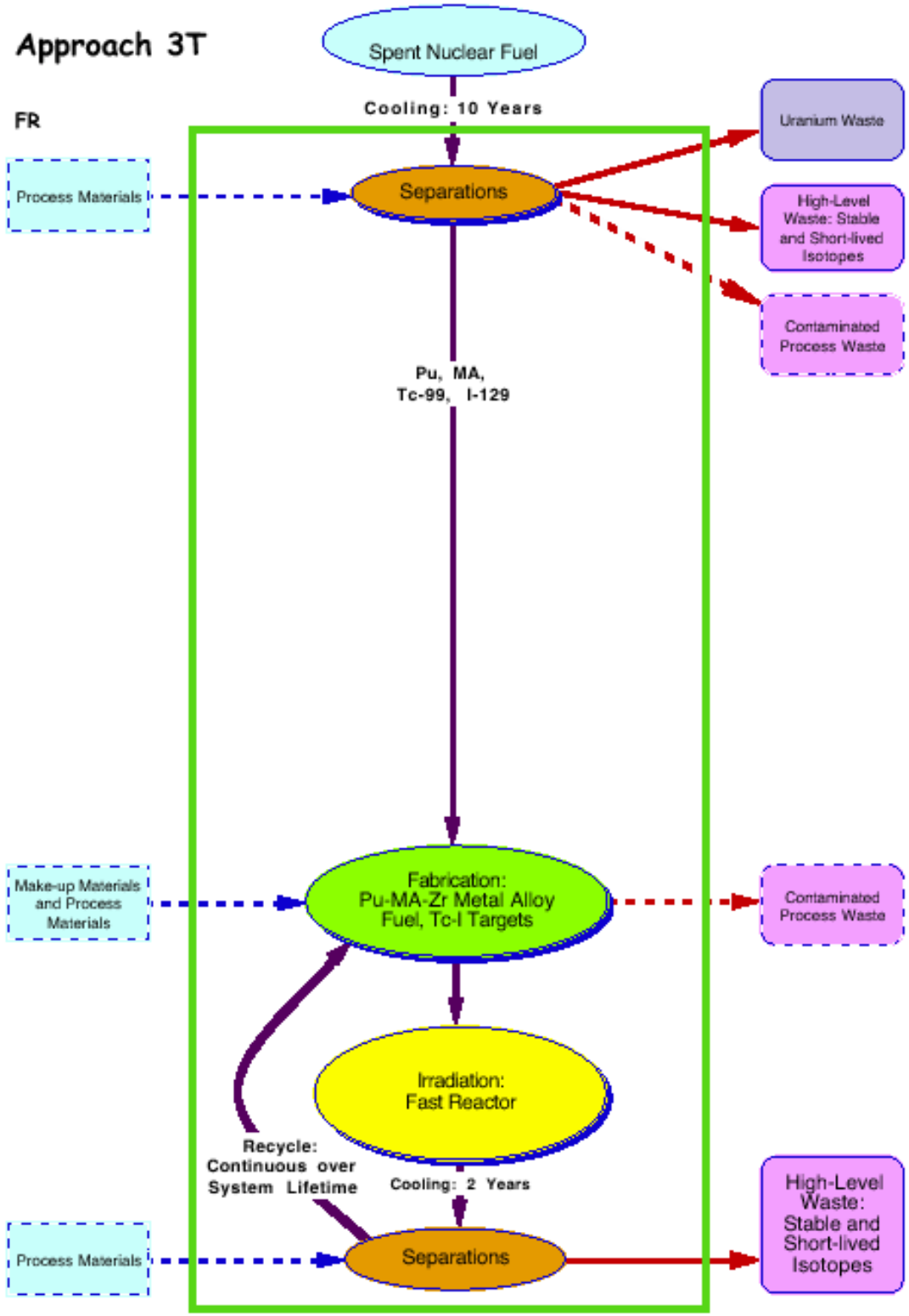


Figure G-10. Flow chart for Approach 3T, the second of the two single-tier approaches. This approach employs a fast reactor rather than an accelerator-driven system.

Appendix H: Quantitative Results for Approaches

Systems Study Task Force

Summary of Mass Flow Results for the Double-Tier Approaches

Table H-1. Tier 1 Performance Results

Parameter	Approaches						
	Pu Separation: MA Bypasses Tier 1				No Pu Separation		
	1X-Stage 1	1X-Stage 2	1Z	1G	2X	2Z	2G
Reactor thermal power (MWt)	3000	3000	3000	600	3000	3000	600
Total number of fuel assemblies	177	177	177	108	177	177	108
Number of fuel batches	3	3	3	1	3	3	1
Cycle length, efpd	443.5	443.5	436.8	500.0	443.5	436.8	400.0
Fuel form	MOX	MOX	NFF	TRISO	MOX	NFF	TRISO
Beginning-of-cycle heavy-metal inventory (t)	76.94	76.94	6.38	0.61	76.94	6.38	0.61
Charge per batch							
Heavy metal (HM) (t)	26.10	26.10	2.57	0.61	26.10	2.57	0.61
TRU in HM (%)	8.85	13.59	100.00	100.00	18.25	99.68	99.70
Pu in HM (%)	8.85	13.59	100.00	100.00	15.81	86.34	86.34
Fissile in HM (%)	6.33	7.69	64.37	64.37	10.76	55.64	55.64
Consumption per batch							
Heavy-metal (%)	5.17	5.17	51.56	57.4	5.18	51.58	45.7
TRU (%)	21.65	16.38	51.64	57.4	13.87	51.80	46.0
Pu (%)	29.66	22.32	56.24	60.7	13.88	54.57	50.3
Fissile (%)	43.33	30.96	74.02	84.6	27.01	78.26	78.1
Discharge burnup (MWd/kg)	51	51	510	591	51	510	470
Post-irradiation cooling period (yr)	7	2	2	-	2	2	-
Average linear power (W/cm)	230	230	230	32.4	230	230	32.4
Burnup reactivity loss (%delta-k)	6.73	5.77	5.42	33.9	3.46	7.95	25.0

*TRISO fuel form is TRU-$O_{1.7}$
NFF form is ZrO_2-$TRUO_2$-Er_2O_3

Table H-2. Performance Results for Tier 2 (Fast Neutron Spectrum System)

Parameter		Approaches								
		Approach 1X	**Approach 1Z**	**Approach 1G**	**Approach 1XT**	**Approach 2X**	**Approach 2Z**	**Approach 2G**	**Approach 3M**	**Approach 3T**
Reactor total power (MWt)		840	840	840	840	840	840	840	840	840
Capacity factor (%)		75	75	75	85	75	75	75	75	85
Cycle length, efpd		140	140	140	310	140	140	140	140	310
Number of fuel batches		7/8	7/8	7/8	7	7/8	7/8	7/8	7/8	7
Number of enrichment zones		2	2	2	2	2	2	2	2	2
Enrichment split		1.30	1.30	1.30	1.25	1.30	1.30	1.30	1.30	1.25
TRU in charge heavy metal (%)		98.7	98.5	98.6	38.3	99.0	97.4	97.5	98.5	32.4
BOEC heavy metal inventory (kg)		3401	3662	4070	13807	2825	3555	3666	2709	13894
BOEC fissile fraction in HM, %		18.7	15.5	11.9	11.9	26.5	15.3	14.4	30.3	13.9
BOEC TRU inventory (kg)		3353	3604	4011	5158	2793	3456	3568	2664	4359
Equilibrium cycle HM Feed (kg)	Recycle	377	411	463	1824	304	397	411	289	1837
	External	119	119	119	266	119	119	119	119	266
Multiplication factor	BOEC	0.970	0.970	0.970	1.019	0.970	0.970	0.970	0.969	1.023
	EOEC	0.942	0.945	0.949	1.001	0.930	0.941	0.943	0.928	1.000
Burnup reactivity loss (%Δk)		2.77	2.53	2.09	1.84	3.93	2.91	2.71	4.14	2.34
Peak linear power (W/cm)		363	360	355	260	380	366	361	385	270
Average discharge burnup (MWd/kg)		223	208	190	119	263	214	209	273	118
Peak fast fluence (10^{23} n/cm^2)		3.73	3.73	3.74	3.97	3.73	3.75	3.73	3.73	3.71
Fraction LWR discharge mass in Tier 2		0.713	0.552	0.501	0.713	0.859	0.484	0.543	1.000	1.000
Average batch heavy metal consumption (%)		23.8	22.3	20.3	12.5	28.0	22.9	22.3	29.0	12.5
Average batch TRU consumption (%)		24.0	22.5	20.5	17.5	28.2	23.2	22.6	29.2	18.6
TRU loss factor (# of cycles)		3.164	3.447	3.884	4.711	2.550	3.317	3.430	2.419	4.364
Post-irradiation cooling period (yr)		2	2	2	2	2	2	2	2	2

Table H-3. Tier 2 Equilibrium Cycle Isotopic Vectors

Heavy Metal Nuclide	Approaches																	
	Pu Separation: MA Bypasses Tier 1								No Pu Separation: Pu+MA in Tier 1						No Tier 1			
	1X: MOX Fuel Form in Tier 1		1Z: NFF Fuel Form in Tier 1		1G: GT-MHR in Tier 1		1XT: MOX Fertile in Tier 2		2X: MOX Fuel Form in Tier 1		2Z: NFF Fuel Form in Tier 1		2G: GT-MHR in Tier 1		3M: TRU Feed		3T: Fertile Feed	
	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)	Charge (w/o)	Dischg. (w/o)
U234	0.911	1.039	1.014	1.116	0.954	1.035	0.483	0.509	0.679	0.810	1.196	1.280	1.014	1.102	0.554	0.669	0.273	0.286
U235	0.201	0.263	0.230	0.287	0.207	0.258	0.177	0.155	0.155	0.214	0.280	0.345	0.224	0.285	0.129	0.179	0.146	0.112
U236	0.219	0.281	0.264	0.327	0.237	0.289	0.338	0.345	0.189	0.254	0.307	0.383	0.260	0.323	0.169	0.229	0.329	0.331
U238	0.001	0.001	0.001	0.002	0.002	0.002	60.692	62.856	0.001	0.001	0.855	0.966	1.044	1.174	0.673	0.816	66.883	69.157
Np237	5.601	3.943	5.887	4.120	6.035	4.196	1.988	1.409	2.829	1.963	2.439	1.683	3.656	2.488	3.794	2.639	1.117	0.809
Pu236	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Pu238	8.078	9.283	7.905	8.802	7.446	8.226	2.477	2.584	7.478	7.370	9.378	7.958	8.681	7.874	5.582	6.505	1.443	1.453
Pu239	11.426	8.435	6.656	5.410	3.330	3.252	9.567	9.284	20.095	13.356	5.890	4.967	5.175	4.507	26.005	16.897	12.858	11.300
Pu240	27.545	28.407	28.705	28.640	27.616	26.834	10.547	10.164	33.689	36.351	28.802	28.611	31.874	31.197	33.172	36.963	9.942	9.731
Pu241	6.347	5.843	7.359	6.278	6.428	5.600	1.905	1.562	7.457	7.433	7.542	6.378	7.562	6.658	6.592	7.152	1.519	1.388
Pu242	15.241	17.118	18.186	19.810	23.699	24.751	4.548	4.487	12.629	15.010	21.452	22.891	21.866	23.218	10.829	13.194	2.477	2.487
Am241	8.890	6.720	7.322	5.692	6.793	5.226	3.176	2.376	4.457	3.761	3.673	3.130	3.649	3.138	4.510	3.772	1.456	1.209
Am242m	0.486	0.528	0.409	0.439	0.380	0.401	0.184	0.202	0.277	0.296	0.219	0.235	0.220	0.235	0.248	0.290	0.089	0.097
Am243	7.235	7.674	7.638	8.196	7.400	8.424	2.243	2.107	4.455	5.369	7.721	8.595	5.437	6.844	3.675	4.565	0.865	0.885
Cm242	0.021	0.584	0.019	0.481	0.017	0.425	0.004	0.085	0.015	0.372	0.015	0.284	0.021	0.280	0.012	0.376	0.002	0.045
Cm243	0.046	0.058	0.040	0.047	0.033	0.040	0.007	0.007	0.035	0.043	0.042	0.038	0.037	0.035	0.028	0.040	0.003	0.004
Cm244	5.334	6.756	5.780	7.132	6.619	7.610	1.207	1.358	3.725	4.969	6.944	8.262	6.456	7.134	2.726	3.909	0.435	0.519
Cm245	1.501	1.868	1.607	1.963	1.721	2.080	0.314	0.341	1.124	1.441	2.007	2.393	1.673	2.050	0.783	1.080	0.110	0.125
Cm246	0.806	1.054	0.863	1.107	0.958	1.193	0.143	0.162	0.618	0.854	1.083	1.399	1.009	1.276	0.449	0.629	0.054	0.061
Fissile Cont. %	20.0	17.0	16.3	14.4	12.1	11.6	12.2	11.6	29.1	22.8	16.0	14.4	14.9	13.8	33.8	25.6	14.7	13.0
HM Consumed,%		23.8		22.3		20.3		12.5		28.0		22.9		22.3		29.0		12.5

Table H-4. Tier 2 External Make-Up Feed Based on Initial 100 kg Heavy Metal from Commercial LWR

Nuclide	PWR discharge (w/o)	Pu Separation: MA Bypasses Tier 1				No Pu Separation		
		1X: MOX	1Z: NFF	1G: GT-MHR	1XT: MOX	2a: MOX	2b: NFF	2G: GT-MHR
U234	0.000		0.077	0.047			0.205	0.119
U235	0.002		0.014	0.003			0.027	0.004
U236	0.002		0.011	0.006			0.013	0.008
U238	0.325		0.000	0.000			0.232	0.322
Np237	6.641	7.768	6.664	6.643	7.768	4.342	2.405	4.202
Pu236	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Pu238	2.749	2.164	2.081	1.710	2.164	6.235	6.698	6.032
Pu239	48.652	15.054	6.092	1.821	15.054	32.361	4.355	4.081
Pu240	22.980	16.449	14.932	14.229	16.449	22.069	13.212	17.560
Pu241	6.926	7.047	7.415	5.996	7.047	8.119	6.653	7.137
Pu242	5.033	6.617	6.925	9.851	6.617	5.558	8.032	9.330
Am241	4.654	10.054	6.010	5.430	10.054	3.761	1.605	1.686
Am242m	0.019	0.051	0.029	0.021	0.051	0.083	0.015	0.017
Am243	1.472	4.152	3.134	1.671	4.152	1.785	2.295	0.249
Cm242	0.000	0.005	0.009	0.007	0.005	0.016	0.025	0.061
Cm243	0.005	0.014	0.013	0.008	0.014	0.020	0.031	0.028
Cm244	0.496	1.672	1.587	2.500	1.672	1.239	2.213	3.276
Cm245	0.038	0.212	0.185	0.138	0.212	0.245	0.323	0.175
Cm246	0.006	0.013	0.014	0.018	0.013	0.016	0.062	0.041
Fissile	55.642	22.380	13.752	7.988	22.380	40.832	11.420	11.442
Total	100.000	71.273	55.193	50.100	71.273	85.850	48.404	54.329

Table H-5. Tier 1 Discharge Isotopic Vector after Two-Year Cooling

Nuclide	PWR discharge (w/o)	Pu Separation: MA Bypasses in Tier 1				No Pu Separation		
		1X: MOX	1Z: NFF	1G: GT-MHR	1XT: MOX	2X: MOX	2Z: NFF	2G: GT-MHR
U234	0.000	0.000	0.184	0.129	0.000	0.000	0.423	0.219
U235	0.002	0.000	0.034	0.007	0.000	0.000	0.056	0.008
U236	0.002	0.000	0.026	0.016	0.000	0.000	0.027	0.015
U238	0.325	0.000	0.000	0.000	0.000	0.000	0.480	0.593
NP237	6.641	0.655	0.002	0.005	0.655	5.059	4.975	7.734
PU236	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
PU238	2.749	4.248	4.977	4.652	4.248	7.263	13.855	11.103
PU239	48.652	29.549	14.571	4.952	29.549	37.704	9.010	7.512
PU240	22.980	32.286	35.710	38.697	32.286	25.712	27.333	32.321
PU241	6.926	13.831	17.737	16.308	13.831	9.456	13.763	13.137
PU242	5.033	12.987	16.566	26.792	12.987	6.476	16.616	17.173
AM241	4.654	2.973	3.208	2.110	2.973	4.382	3.319	3.103
AM242M	0.019	0.032	0.024	0.007	0.032	0.097	0.032	0.031
AM243	1.472	2.291	3.964	0.543	2.291	2.079	4.747	0.459
CM242	0.000	0.010	0.022	0.020	0.010	0.019	0.052	0.111
CM243	0.005	0.008	0.017	0.007	0.008	0.024	0.065	0.052
CM244	0.496	0.999	2.604	5.448	0.999	1.444	4.579	6.030
CM245	0.038	0.131	0.352	0.273	0.131	0.285	0.669	0.322
CM246	0.006	0.000	0.000	0.032	0.000	0.000	0.000	0.076
Fissile	55.642	43.551	32.736	21.555	43.551	47.565	23.594	21.060
Total	100.000	100.000	100.000	100.000	100.000	100.000	100.000	100.000

Table H-6. Spontaneous-Fission Neutron Source (n/s-g)

Nuclide	ORIGEN-RA
U234	5.654E-03
U235	2.993E-04
U236	5.489E-03
U237	1.659E-06
U238	1.362E-02
U239	1.274E-05
Np237	1.046E-04
Np238	5.455E-05
Np239	1.804E-03
PU236	3.394E+04
PU238	2.588E+03
PU239	2.182E-02
PU240	9.087E+02
Pu241	4.941E-02
PU242	1.718E+03
PU244	1.900E+03
AM241	1.187E+00
AM242M	1.347E+02
AM243	3.933E+00
CM242	2.100E+07
CM243	1.221E+03
CM244	1.080E+07
CM245	3.875E+01
CM246	9.448E+06
CM248	3.925E+07
CM250	1.006E+09
CF250	1.132E+10
CF252	2.315E+12
CF254	1.221E+15

Appendix I: Assumptions and Flow Charts for Separations

The material flows for the LWR, Tier 1, and Tier 2 fuel separations calculations were obtained from the detailed core-performance physics calculations for the nine distinct approaches, and from a uniform set of separations assumptions. For any one approach, the basis for the material flows was 1 metric ton of TRU in the 10-year cooled PWR fuel, or a total spent LWR fuel rate of 101 metric tons. The recovery factors assumed in the calculations are summarized in Table I-1.

Table I-1. Recovery Factors used in Separations Calculations

Parameter	Solvent Extraction	Pyroprocessing
U recovery	**99.9 wt%**	**99.9 wt%**
Disposition of U losses	.05 wt% in metal waste .05 wt% in glass	.05 wt% in MWF* .05 wt% in CWF**
TRU recovery	**99.9 wt%**	**99.9 wt%**
Disposition of TRU losses	.05 wt% in metal waste .05 wt% in glass	.05 wt% in MWF .05 wt% in CWF
I recovery	**95 wt%**	**95 wt%**
Disposition of I losses	5 wt% to gas	5 wt% to gas
Tc recovery	**95 wt%**	**95 wt%**
Disposition of Tc losses	.05 wt% in metal waste 4.95 wt% in glass	5 wt% in MWF

* MWF = metal waste form.
**CWF = ceramic waste form.

Recovery of uranium and plutonium at 99.9% has been demonstrated commercially with solvent-extraction technology. The 99.9% recovery of minor actinides by solvent extraction or of U, Pu, and minor actinides by pyrochemical processing is a project goal not yet demonstrated at the industrial scale but derived from experience at lesser scales. In some approaches, achievement of the project recovery goals will require additional engineering advancements.

The adoption of the 99.9% recovery factor for all separations eliminates any bias that might arise from the use of other speculative values. For any test approach, the required separations may be achievable by different technologies. Table I-2 details some of the options. The selection of a specific technology option would require a more in-depth analysis of system requirements, including product quality, capacity, and economics.

Solvent extraction yields very pure U and I or Pu. The material-flow calculations assume no impurities. The U product from solvent extraction of LWR fuel is assumed to meet Class C disposal requirements. The separations achievable by pyroprocessing are not expected to be as selective. In pyroprocessing of LWR fuel (Approach 3T), the uranium product is assumed to have 50 ppm of residual TRUs. For this approach, the uranium product will ultimately be consumed in the reactor and not discharged as waste.

Rare earth fission products are not completely separated from TRUs in pyroprocessing. The calculations assume that 5 wt% of the rare earths in the fuel remain with the TRU product. Rare earth fission products are not easily separated from TRUs (specifically Am and Cm) via solvent extraction either. An extra extraction step must be added to the process for the separation. For the purposes of these calculations (Approaches 2X, 2Z, and 2G), the minor actinides are assumed to be recovered without rare earth contamination in this added solvent extraction step.

In other words, the TRU feed to Tier 1 reactors was assumed to be pure, containing no rare earths.

Several assumptions are embodied in the estimates of waste generation tonnage and volume. The activation and fission products separated by solvent extraction are disposed in two waste types: metal waste and glass. The metal waste contains the hulls (activation products) and a small amount (.05wt%) of residual fuel (U, TRUs and fission products). The remaining active metal and noble metal fission products are disposed as glass. The rare earth fission products are either separated with the minor actinides (Approaches 1X, 1Z, and 1G) as product or separated from the minor actinides (Approaches 2X, 2Z, and 2G) and disposed with the glass.

The metal waste from commercial solvent extraction (e.g., COGEMA) is typically disposed of as is or crushed for volume reduction. ANL has developed a metal waste form (MWF) technology that could be effectively applied to this waste. The metals in the fuel form the alloy: either 85% stainless steel and 15 wt% Zr (SS-15Zr) or 92% Zr and 8% stainless steel (Zr-8SS). In the calculations, the weight of the metal waste is assumed to be equivalent to the weight of the fuel hardware, cladding and activation products. The volume is calculated assuming a 7.6 g/cc density of the MWF.

The glass waste generated from commercial aqueous processing contains 20% to 25% fission-product oxides (that have been cooled for >10 years). The weight and volume of the glass were estimated as:

- Metric tons glass = (metric tons fuel components, metal basis, in glass) * 8.7
- Cubic meters glass = (metric tons fuel components, metal basis, in glass) * 3

For the test approaches, however, we found that the heat content of the glass at this commercial dilution was excessively high; hence, the weight and volume were recalculated to yield a maximum heat load of 2,000 watts/cubic meter.

In pyroprocessing of LWR and metal fuels, the activation products and fission products are discharged in two wastes: MWF and ceramic waste form (CWF). The cladding, activation products, noble metal fission products, and a small fraction (.05wt%) of fuel residuals are disposed of in the MWF. The weight and volume of MWF were calculated (as described above) for aqueous processing.

The active metal and rare earth fission products are disposed of with process salt (LiCl or LiCl/KCl) in the CWF. The weight and volume of the CWF can be estimated as:

- Metric tons CWF = (metric tons fuel components, metal basis, in CWF) * 64
- Cubic meters CWF = (metric tons fuel components, metal basis, in CWF)* 30

As with the glass, however, the heat content of the CWF at this chemical dilution was excessively high. The weight and volume were recalculated to yield a maximum heat load of 2,000 watts/ cubic meter.

The pyroprocessing of the gas reactor fuels involved fluoride volatility in fluoride salts. For this approach, all the fission products were disposed as a fluorapatite. The weight and volume of the fluorapatite at chemical dilution were calculated as shown below and then adjusted to a maximum heat load of 2,000 watts per cubic meter.

- Metric tons fluorapatite = (metric tons fuel components in fluorapatite) * 40
- Cubic meters fluorapatite = metric tons fluorapatite / 3.2

Significant quantities of C and Si from the GA fuel coating and rod graphite must also be disposed. In the fluoride volatility process, these species are fluorinated and volatilized. The gases (CF_4 and SiF_4) are reacted with lime (CaO) to form a solid for disposal:

- $CF_4 + 3CaO \rightarrow CaCO_3 + 2CaF_2$
- $SiF_4 + 2CaO \rightarrow SiO_2 + 2CaF_2$

At this point, we do not know if these solids must be further stabilized for disposal. Assuming not, the weight and volume of this waste stream were estimated:

- Metric tons C+Si waste = (metric tons C+Si) * 21.3
- Cubic meters C+ Si waste = (metric tons C+Si waste) / 2.5

The impact of heat load on waste volumes needs to be highlighted. The waste volumes calculated are an artifice of the cool-down assumed. The Systems Working Group selected a two-year spent fuel cool-down for all test approaches except Approach 1X, Tier 1, Cycle 1. The fission products that contribute to the heat load of the waste (and hence increase its volume over that required by chemistry) cool by an order of magnitude in five to 10 years; therefore, increasing the cool-down would substantially reduce the volume of waste, as well as increase the engineering viability of processing and handling the spent fuel, waste, and products.

Note: To simplify the calculations, the fission products were subdivided into groups as shown in Table I-3 to define their chemistry or separation behavior. Separation calculations were performed on the sum of the components in the group, not on individual components. For example, if 5 wt% of rare earths were distributed to the TRU product in a separation step, that 5 wt% would have the same composition as the rare earths in the spent fuel; therefore, even though erbia may be more likely to partition than Ce to the TRU product, both were assumed to partition to the same extent.

Table I-2. Fission Product Groupings for Separations Calculations

Group	Fission products
Gases	H, He, Kr, Br, Xe, I
Active Metals	Rb, Y, Sr, Cs, Ba
Rare Earths	La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er
Noble Metals	Zn, Ga, Ge, As, Se, Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Te

Table I-3. Separations Technology Options

Process Name	Type	Potential Application	Technology Status	Comments
UREX	Aqueous solvent extraction process for U and Tc recovery	Commercial SNF treatment (all); Tier 1 recycle (Approach 1X); Tier 1 discharge (Approaches 1X, 2X)	Laboratory-scale; Engineering-scale demo planned for FY02	Class-C U product and highly effective for Tc recovery
PUREX	Aqueous solvent extraction process for U and Pu	Commercial SNF treatment (all); Tier 1 recycle (Approach 1X); Tier 1 discharge (Approaches 1X, 2X)	Commercial process at >1000 MTHM/y	Class-C U product, modifications required for Tc recovery
TRUEX	Aqueous solvent extraction process for TRU recovery	Commercial SNF treatment (all); Tier 1 recycle (Approach 1X); Tier 1 discharge (Approaches 1X, 2X)	Pilot-scale process, not applied commercially	Effective means to partition and recover any two of the following Np, Pu, and/or Am/Cm/RE
DIAMEX	Aqueous solvent extraction process for TRU recovery	Commercial SNF treatment (all); Tier 1 recycle (Approach 1X); Tier 1 discharge (Approaches 1X, 2X)	Laboratory-scale R&D hot testing	Effective means to partition and recover any two of the following Np, Pu, and/or Am/Cm/RE
SANEX	Aqueous solvent extraction process for Am and Cm recovery	Commercial SNF treatment (all); Tier 1 discharge (Approaches 1X, 2X);	Laboratory-scale R&D hot testing	Novel process for separating Am and Cm from rare earths
Oxide reduction	Pyrochemical process for converting metal oxides to metals	Commercial SNF (MA and FP approach 1); Commercial SNF (Approaches 2Z, 3T, 3M); Tier 1 discharge (all except 1G, 2G)	Laboratory-scale development testing	Newly-developed process, produces less waste than traditional reduction methods
Electro-refining	Pyrochemical process for partitioning and recovering actinides	Commercial SNF (MA and FP approach 1); Commercial SNF (Approaches 2Z, 3T, 3M); Tier 1 discharge (all except 1G, 2G); Tier 2 recycle (all)	Engineering-scale hot testing with EBR-II spent fuel, laboratory-scale testing for TRU recovery	Effective process to partition and recover actinides from short-cooled ADS and ALMR fuels
Electrolysis	Pyrochemical process for actinide recovery	Commercial SNF (MA and FP approach 1); Commercial SNF (Approaches 2Z, 3T, 3M); Tier 1 discharge (all); Tier 2 recycle (all)	Laboratory-scale development testing	Potentially effective and efficient means for actinide recovery from molten salts
Fluoride volatility	Volatility process for treating graphite-coated particle fuels	Tier 1 Discharge (Approaches 1G and 2G)	Conceptual flow sheet development stage, fluoride volatility demonstrated on other fuel types at engineering-scale	Potentially effective method for removing coating and converting oxide fuel kernels to soluble salts for pyrochemical processing

Appendix J: Assumptions and Flow Charts for Fuel Fabrication and Utilization

This appendix documents the assumptions used for this multi-tier evaluation regarding the recommended fuel type (and candidate options) for each system, fuel form compositions, and estimated burnup limits. Tier 1 options are considered to be power producing thermal reactor systems. This could involve the use of some of the currently operating commercial LWRs, newly constructed advanced LWRs, and newly constructed gas-cooled reactors. The two basic fuel types considered for this tier are solid pellet fuel (with the same geometry as current commercial UO_2 fuel) and particle fuel (similar to particle used in the GT-MHR designs). The compositions include either separated PuO_2 or PuO_2 with MA oxide blends in either fertile (mixed with depleted UO_2) or nonfertile mixes (inert matrix material such as ZrO_2). For Tier 2 only, fast-spectrum systems are being considered with a variety of fuel forms.

For the current LWRs and ALWR systems, both fertile and nonfertile fuel forms are considered. The fertile fuel forms, considered in Approaches 1X and 2X, are a Pu mixed-oxide and a Pu-MA mixed-oxide, respectively. For the nonfertile fuel, the proposed fuel form is Pu or TRU (Pu and MA) oxides embedded in an inert ZrO_2 matrix (denoted Z, in Approaches 1Z and 2Z, respectively). The latter two approaches have also been investigated at the Paul Scherrer Institute in Switzerland. The presence of minor actinides in Approaches 2X and 2Z will cause handling and performance (helium generation) problems. Remote fabrication of MOX fuel has been performed at facilities such as Melox in Marcoule, France; however, facility maintenance may still require some hands-on access.

For the gas-cooled systems, a nonfertile TRISO particle fuel is proposed, based on the GT-MHR designs that have been proposed for transmutation. For Approach 1G, the potential inclusion of the neptunium with the plutonium would reduce the amount of material directly charged to the second tier, but would also limit the neutrons available for long-lived fission product (LLFP) transmutation. For Approach 2G, the fuel will generate more decay heat when the americium and curium isotopes are included; and the helium generated by these isotopes may be a major issue. A different fuel kernel may be considered, such as an oxycarbide that might offer enhanced stability.

For the Tier 2 fast-spectrum systems, a variety of fuel forms are being considered. For Approach 3T, a fertile fuel form based on the ternary metal alloy (U-Pu-10Zr, denoted t) developed in the Integral Fast Reactor Program is proposed; for that approach, the inclusion of uranium in the fuel and the core designs envisioned would provide for a TRU conversion ratio of ~0.5. For all other fast-spectrum systems, only nonfertile fuel forms (which provide the maximum possible destruction rate) are evaluated. The nonfertile fuel forms currently under development are a TRU-Zr metal alloy (denoted m), a mixed nitride pellet (TRU, Zr)N (denoted n), a mixed oxide pellet (TRU, Zr)O_2 (denoted o), and a nitride-particle fuel dispersion in a metal Zr matrix (denoted d). For all approaches considered, however, the metal alloy (with a Zr content of 40 wt.%) is assumed for the evaluation. This selection is not based on any preference for that fuel form, but rather is caused by the fact that neutronic models for that fuel form are more readily available to support the multi-tier evaluation. Characteristics that are unique to the metal fuel form do lead to discrimination between approaches considered in the present analysis.

Calculations of expected nuclear performance in the transmuter are performed with the assumption of a limit on peak fast fluence on fuel cladding and assemblies (fabricated using the stainless steel HT9) of 4×10^{23} n/cm^2 (fast fluence, or neutrons with E>0.1 MeV). This value is derived from the existing irradiation performance database, which includes irradiation of HT9-clad

MOX fuel with HT9 ducts in the Fast-Flux Test Facility (FFTF), which attained fluences of 3.9×10^{23} n/cm^2, with no observed performance concerns. (See, for example, Leggett and Walters [1], or Baker, Bard, Leggett, and Pitner [2].) When transmuter cycle calculations are performed, the peak fuel burnup value that corresponds to the stated fluence limit is determined and assumed as an attainable peak burnup limit. These values range from 30 atom% to 40 atom%.

However, it should be noted that the calculated burnup values are comparable, in terms of number of fissions per unit volume of fuel material, to a smaller burnup value in the more familiar MOX or metallic fast reactor fuels due to the lower heavy metal density of the proposed nonfertile fuels. For example, 40 atom% burnup in a TRU-40Zr fuel would provide the same density of fissioned atoms as roughly 15 atom% burnup in U-20Pu-10Zr. Therefore, the amount of fission products generated in the proposed fuel designs after exposure to fast fluence of 4×10^{23} n/cm^2 will be less than that attained in MOX and U-Pu-Zr. However, the burnup values that can be attained in the proposed fuel designs are not yet known, and will be affected by other phenomena suspected to be more severe in nonfertile fuels, such as helium generation and release, fuel-cladding chemical interaction, or swelling. Ultimately, burnup limits will be determined through experimentation under a variety of steady state and transient conditions and verified through a fuel qualification program.

In general, the burnup limit will degrade at higher contents of the minor actinides; therefore, special design features will likely be incorporated to allow high burnup in the second-tier systems *(because minor actinide contents increase during irradiation in the first tier*); this might include cladding liners, large plenum spaces, and low smear density. Gas release (as opposed to swelling) will likely be a positive attribute because it will alleviate fission gas-induced swelling. Gas release can be promoted and accommodated in the metal and nitride pellet forms, which allow for design or evolution of higher porosity and for larger fuel-cladding gaps, but is especially difficult to accommodate in a dispersion fuel that has historically been designed for no gas release. The dispersion fuel remains an option, however, because a dispersion fuel fabricated with low-volume loading has the potential to achieve particularly high burnup. For Approach 1 options (1X, 1Z, 1G), the minor actinide content in the second tier will be quite high and helium generation resulting from neutron capture in americium will likely influence fuel behavior and lifetime.

It must be noted that eventual selection of fuel separations technology options that are currently being developed for transmutation will have some influence over the selection of a reference fuel form, and *vice versa*. Clearly, compatibility with the separations technology will be important, and optimization of fuel and transmuter performance considerations, along with those for separations efficiency and cost, will be incorporated into eventual technology selection.

Regarding fabrication issues, remote fabrication has been demonstrated for metal fuel in the US and nitride fuel in Russia. The nitride fuel options will require N-15 enrichment and recovery in the recycle process, which has not been demonstrated in a remote process. In general, the metal fuel fabrication will be simplest and easiest to maintain, with the mononitride fuel likely easier than mixed valence oxide forms. The TRISO fuel will be more difficult to fabricate than oxide (additional steps), but it offers superior burnup potential. The dispersion fuel will be more difficult to fabricate than metal or pellet options but it may have an advantage over pellet options because the process eliminates some of the dusting concerns. The volatility of americium and its compounds may complicate the envisioned fabrication processes for these fuel forms, however; this issue is expected to be a greater concern for the metal-alloy fuel than for the ceramic fuel forms.

Based on previous technical experience with each of the fuel forms, primarily with nonfertile analogues used in or developed for analogous reactor systems, design parameters and

performance could be determined or assumed for the purposes of this multi-tier analysis. This data is summarized in Tables J-1 through J-3.

Table J-1. GT-MHR Fuel Composition and Estimated Burnup Limits Assumed for the Multi-Tier Analysis

Component	Composition	Dimension (micrometers)	Density (g/cc)
Kernel	$PuO_{1.7}$ (for high MA approaches PC_xO_y - x and y to be determined, x probably about 0.5)	200 (diameter)	10.4
Low-density pyrocarbon	Carbon	100 (thickness)	1
Inner pyrocarbon	Carbon	35 (thickness)	1.8
Silicon carbide	Silicon Carbide	35 (thickness)	3.2
Outer pyrocarbon	Carbon	40 (thickness)	1.8
COMPACT	TRISO particles in carbon of density 0.7 grams/cc TRISO packing fraction maybe 10% to 20% by volume		
Burnup for high Pu (Approach 1) 750,000 MWd/ MT of initial metal; Burnup for high-MA approaches will be lower			

Table J-2. ALWR Fuel Composition and Estimated Burnup Limits Assumed for the Multi-Tier Analysis

	MOX Fertile Fuel	Inert Matrix Nonfertile Fuel*
Estimated composition	DUO_2-(.55 g/cc)PuO_2	ZrO_2-PuO_2-Er_2O_3-Y_2O_3 **
Fuel density	96 %TD (~10.53 g/cc)	7.28 g/cc
Burnup limit (peak)	53 GWd per metric ton	810 GWd per metric ton
Expected net Pu destruction	~30% (includes Pu production)	~70%
Net minor actinide production	5.5% of initial Pu	6.7% of initial Pu

*The nonfertile fuel can be tailor-made with different blends of the burnable poisons to maximize burnup, reduce effects on Doppler and void coefficients, and improve overall efficiency of the Pu burn.

** 80-90 atom% ZrO_2, 7%–14% PuO_2, 3%–6% Er_2O_3 and Y_2O_3 is used as stabilizer for ZrO_2 [3].

Table J-3. Fast-Spectrum System Compositions and Estimated Burnup Limits Assumed for the Multi-Tier Analysis

(Calculations were performed assuming metal-alloy fuel in fast-spectrum systems)

Fuel Type	Approx. Composition[1] wt.%	Approx. TRU density,[4] g·TRU/cm^3	Burnup limit, atom% HM	Comments
Fertile Metal	U-TRU-10Zr	14.15	20[3]	Similar to IFR ternary metal fuel.
Metal	TRU-40Zr[2]	5.8	30	Minor alloying additions may be required to alloy or form compounds with Np.
Nitride[5]	TRUN-30ZrN	5.8	30	85% theoretical density per Blank [5]
Oxide	$TRUO_x$-$16ZrO_2$	5.8	30	85% theoretical density; initial density selected for optimization of gas release and other characteristics
Dispersion	TRUN (coated) in Zr matrix	3.2	30+	25 vol.% particulate loading. May not work at high-MA content due to no gas release.

[1] Compositions are adjusted to maintain a TRU density of 5.8 g TRU/cm^3 for metal, nitride, and oxide.

[2] Composition chosen on the basis of cubic phase stability in Pu-Zr binary system. Americium is a δ-phase stabilizer and may allow more Zr-rich compositions.

[3] A limited number of ternary metal fuel pins were successfully tested in EBR-II and FFTF to approximately 10 atom% burnup and individual pins to 20 atom% burnup; special measures (e.g., lower smear density, high plenum volume, duplex cladding) could increase burnup to 30% range if lifetime is not fluence limited.

[4] Assuming TRU metal density of δ-Pu; 15.4 g/cm^3, TRUN density is 14 g/cm^3, TRUO density is 11 g/cm^3.

[5] Sasa *et al*. [6] claim TRU-based fuel nitrides (90w/o TRU-10w/o Pu) have higher thermal conductivity, higher heavy-metal density, and much higher melting points than equivalent metal fuels. If this is true, we should consider comparing TRU densities for metals and nitrides at least as equal so the inert component (Zr) would be equal as well.

References

1. Leggett, R.D. and L.C. Walters, “Status of LWR Fuel Development in the United States of America,” *Journal of Nuclear Materials*, v. 204, pp. 23–32, September 1993.

1. Baker, R.B., F.E. Bard, R.D. Leggett, and A.L. Pitner, “Status of Fuel, Blanket, and Absorber Testing in the Fast Flux Test Facility,” *Journal of Nuclear Materials*, v. 204, pp. 109–118, September 1993.

3. Kasemeyer U., J.M. Paratte, P. Grimm, and R. Chawla, “Comparison of Pressurized Water Reactor Core Characteristics for 100% Plutonium-Containing Loadings,” *Nuclear Technology*, v. 122, pp. 52–63, April 1998.

4. Blank, H., "Specification and Characterization of Dense Fuels for Liquid-Metal-Cooled Fast-Breeder Reactors," *Journal of Nuclear Materials*, 153, pp. 171–177, 1988.

5. Sasa, T., T. Nishida, T. Takizuka, O. Sato, and N. Yoshizawa, “Neutronics and Burnup Analysis of an Accelerator-Based TRU-Nitride Fuel Transmutation System with the ATRAS Code,” *Progress in Nuclear Energy*, 32, pp.485–90, 1998.

Appendix K: Assumptions and Flow Charts for Neutronic Analyses

Five distinct system types— advanced fast reactor (ALMR), commercial ALWR, transmutation ALWR, GT-MHR, and fast-spectrum ADS (i.e., ATW)—were identified and briefly described in Appendix F. In this appendix, the analytical techniques used to analyze neutron transmutation (i.e., depletion and production), system performance, and detailed spent-fuel characteristics are described. Key assumptions employed in these neutronics analyses are also explicitly identified.

Commercial Advanced Light-Water Reactor

No detailed computations were performed for the commercial ALWR system. Instead the ORIGEN2 computations developed in the Oak Ridge National Laboratory (ORNL) report, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018, were reproduced. The standard-burnup computations for *generic* reactor systems developed in this report were used to estimate the Yucca Mountain Project (YMP) spent-fuel inventory. The standard PWR benchmark composition with extended cooling time (25 years) was also used as the feed material for previous ATW system studies. For all approaches in this study, the extended-burnup PWR benchmark was used with an assumed cooling time of 10 years before the commercial ALWR fuel is processed for insertion into the transmutation system. This feed composition is compared to the YMP inventory in Table K-1.

Table K-1. Commercial Spent-Fuel Feed Specification for AAA System Studies

Isotope	YMP Inventory[a]	ALWR[b]
U-235	0.004	0.002
U-236	0.002	0.002
U-238	0.478	0.325
Np-237	5.023	6.641
Pu-238	1.272	2.749
Pu-239	53.196	48.652
Pu-240	21.534	22.980
Pu-241	3.782	6.926
Pu-242	4.686	5.033
Am-241	8.967	4.654
Am-242m	0.014	0.019
Am243	0.926	1.472
Cm242	0.000	0.000
Cm243	0.002	0.005
Cm244	0.104	0.496
Cm245	0.009	0.038
Cm246	0.001	0.006

[a]Based on the medium burnup (33,000 MWd per metric ton) PWR benchmark with 25 years cooling.

[b]Based on the high burnup (5,000 MWd per metric ton) PWR benchmark with 10 years cooling.

Transmutation Advanced Light-Water Reactor

A quick and accurate modeling path was developed for estimating mass flows in the initial irradiation in ALWR systems. The technique models both standard mixed oxide (MOX) and nonfertile fuel (NFF) options in a consistent manner. A unit lattice (assembly) analysis approach was used. This is considered sufficient because the systems are assumed to employ a full-core loading of a single fuel form (e.g., 100% MOX fuel or NFF), in which case neglect of interassembly spectral interferences are not expected to invalidate calculated mass flows.

A variety of thermal-reactor lattice codes were evaluated for this application, and calculational approximations were analyzed to ensure that representative mass-flow values are calculated. The WIMS8 code was selected at the end of this process because of the availability of a 172-group library (constructed from JEF 2.2 nuclear data library) and the explicit treatment of over 100 fission products (allowing explicit tracking of Tc-99, I-127, and I-129, among others). The code also covers heavy-metal isotopes from U-233 to Cm-245. Preliminary calculations done for an OECD/NEA unit-MOX-cell benchmark problem indicated the need for a 172-group library for approaches containing degraded Pu vector (i.e., with significant amounts of the higher Pu isotopes). The basic WIMS 69-group structure, which is used worldwide, does not provide sufficient detail for the accurate representation of the low-energy resonances of the higher Pu isotopes, particularly Pu-242. Using the 69-group library on the benchmark problem resulted in an underprediction of the cell k-infinity by about 4%, compared to a reference Monte Carlo result obtained with MCNP4 and JEF 2.2 library. The 172-group library gave a k-infinity value that was about 0.3% lower than the MCNP value. It should be noted that the same study indicated the 69-group library gives an accurate representation of the MOX cell, when the predominant Pu isotopes are Pu-239 and Pu-240, because the energy range containing their low-energy resonances are sufficiently covered by the 69-group structure.

Using the WIMS8 model for analyzing the core, it is assumed that the burnup point with a k-infinity of 1.035 corresponds to the LWR end-of-cycle state. The discharge burnup is then estimated by proportionally scaling this unit-lattice burnup point to account for the fuel-loading scheme. In our LWR studies, a three-batch core is assumed. This implies that the end-of-cycle state roughly corresponds to twice-burned fuel; thus, the total fuel lifetime is estimated at 3/2 the interval required to achieve a k-infinity of 1.035. Fission-product and heavy-metal masses at this discharge burnup are additionally modified in a zero-power WIMS8 calculation to account for the cooling interval, which is assumed to be seven years for MOX fuel or two years for the other fuels.

Several sensitivity studies were performed to assess the accuracy of the WIMS8 model. Because the WIMS8 code allows the unit assembly calculation to be performed in various stages, we studied the effect of using the full 172-group structure for pin-cell resonance treatment calculations and the application of a reduced set of group-condensed cross sections (from the cell calculations) in the unit-assembly calculation. This was thought necessary because of the number of approaches to be investigated and the fact that the WIMS8 computation time increases with the number of groups used, particularly for the assembly calculations. Our study indicated that employing a 28-group structure in the assembly-level calculation gives sufficient accuracy. Comparing the 28-group results to the 172-group results, the difference in k-infinity is less than 0.11% and that in the discharge masses following a seven-year cooling interval is less than 1.5% (only the difference in Cm-245 mass is greater than 1%). The computational time for the 28-group approach was a factor of about five lower than that of the 172-group approach.

In WIMS8 depletion calculations, the user also has control over the spectrum frequency and flux-normalization calculations. At the beginning of each *cycle* the space- and energy-dependent flux is calculated; each cycle can be divided into a number of *steps* at which the flux is normalized to

the specified power level. Depletion calculations are performed with the flux at the beginning of the step. The sensitivity of the lattice k-infinity and isotopics to the cycle and step sizes was evaluated in detail. For the reactor-grade MOX fuel multi-tier studies, a calculation cycle-length of 2,000 MWd per metric ton with 10 steps was used; this speeds the computation time by a factor of seven and results in k_{inf} errors of only 0.015% as compared to more detailed cycle/step models. Equivalent values for the nonfertile fuel approaches are 20,000 MWd per metric ton and 10 steps.

Using this approach, a WIMS8 model was developed to evaluate the 100% MOX and NFF full-core problems that were analyzed by the Paul Scherrer Institute (PSI) in *Nuclear Technology*, (1998) to verify that the model can accurately predict the core discharge isotopics. The approaches modeled are those initially loaded with fuel having reactor-grade (RG) isotopics similar to those of interest in our multi-tier studies. In the MOX approach, the initial Pu fraction in the heavy metal is 6.1 w/o. For the NFF approach, an annular pellet fuel containing a central reduced-density ZrO_2 zone and an outer uranium-free fuel zone containing zirconia-based fuel (ZrO_2-PuO_2-Er_2O_3-Y_2O_3) were modeled. A comparison of the discharge isotopics from the WIMS8 and PSI whole-core results is summarized in Table K-2.

Table K-2. Beginning- and End-of-Life (BOL and EOL) Isotopic and Total Concentrations for PSI MOX and NFF ALWR Cores (Relative to Initial Pu Content)

	RG-MOX			RG-NFF		
	BOL (%)	EOL (%)		BOL (%)	EOL (%)	
Quantity		WIMS8 (1330 days)	PSI (1300 days)		WIMS8 (1310 days)	PSI (1300 days)
Am-241	0.00	0.87	0.84	0.00	0.72	0.70
Am-243	0.00	1.99	2.04	0.00	2.27	2.20
Cm-242	0.00	0.24	0.21	0.00	0.26	0.23
Cm-244	0.00	1.28	1.23	0.00	1.60	1.55
Cm-245	0.00	0.15	0.14	0.00	0.20	0.18
Np-237	0.00	0.31	0.41	0.00	0.00	0.00
Pu-238	1.30	1.40	1.30	2.70	2.00	2.00
Pu-239	60.30	24.70	23.90	54.50	6.80	6.50
Pu-240	24.80	21.90	21.70	22.80	15.20	15.10
Pu-241	9.00	12.30	12.30	11.70	9.00	9.10
Pu-242	4.60	7.70	7.60	8.30	10.20	10.20
Total Pu	100.00	67.90	66.80	100.00	43.30	42.90
Total MA	0.00	5.00	5.02	0.00	5.07	4.99

Based on these results, the WIMS8 models appear to give an accurate tracking of the assembly reactivity and TRU isotopics, so the WIMS8 lattice computations were used to generate the core discharge masses for the various ALWR approaches in the multi-tier scenarios. The mixed-oxide fuel used in the study is UO_2-PuO_2, while the nonfertile fuel is ZrO_2-$TRUO_2$-Er_2O_3; the ZrO_2 matrix would probably have to be stabilized using an additive like yttria (Y_2O_3), which has a low neutron cross-section. For all the ALWR approaches, fuel assemblies are assumed loaded in three batches and discharged at a reactivity-limited burnup of 51 GWd per metric ton for the full-MOX approaches and 510 GWd per metric ton for the full-NFF approaches (corresponding to a core residence time of 4.5 years at about 80% capacity). By assuming a core neutron leakage of 3.5% Δk, the fresh fissile loading needed to meet the cycle specifications was derived by adjusting the TRU loading (%TRU/HM) in the MOX-fueled assembly such that the end of cycle kinf = 1.035. The EOC state was approximated in the lattice calculation by unpoisoned conditions (i.e., 0 ppm

soluble boron) at two-thirds the discharge burnup (the core averaged burnup at EOC). This approach was also used for the NFF approaches, except that in these approaches the erbia (Er_2O_3) weight fraction in the fuel was varied, and compensated by the zirconia (ZrO_2) amount; the heavy-metal weight fraction in the NFF could also be varied and has been in some of the searches.

Several additional assumptions were employed for the ALWR MOX fuel calculations. Soluble boron is typically used for global reactivity control to maintain criticality during the operation cycle of an LWR. In a UOX-fueled core, the critical soluble boron concentration is ~1,000 ppm at beginning of cycle and is gradually reduced to 0 ppm at EOC, giving a cycle-averaged soluble boron concentration of ~500 ppm. Nonetheless, soluble boron was neglected in this analysis. To accurately account for the impact of soluble boron would have meant repeating all calculations with an appropriate cycle-averaged soluble boron concentration once the BOL TRU loading had been determined. The cycle-averaged soluble boron concentration that would be needed for this additional set of calculations is also dependent on the lattice design and fuel loading, requiring yet another set of calculations for each approach. Given the short turn-around time demanded for these multi-strata studies, it was decided to forgo these detailed calculations at present; however, a few approaches were evaluated to assess the impact of neglecting the soluble boron on the results presented here. Even in the extreme case of a cycle-averaged boron concentration of 1,000 ppm, the impact of soluble boron on the discharge TRU masses is minor; the total TRU mass at discharge is affected by only 1%, so it was concluded that the impact on transmutation performance would be negligible.

In Approach 1, where the plutonium is separated from the commercial ALWR fuel, the Pu makes two passes through the MOX ALWR tier, with an intermediate PUREX processing step between Stage 1 and Stage 2 to remove the minor actinides (which are diverted to the second-tier system) and fission products. To meet the same discharge burnup constraints in both stages, the second stage requires a 50% increase in TRU loading because of the degradation of the Pu isotopic vector that occurs during the first stage of irradiation. PUREX-processing is assumed for the MOX fuel approach; a seven-year cooling and refabrication interval is used to reflect typical five-year cooling requirements for PUREX. For all approaches, a two-year interval was assumed for pyro-processing of material passing to the fast-spectrum tier.

For the nonfertile fuel approaches, a discharge burnup of 510 GWd per metric ton is used, about a factor of 10 more than that of the mixed-oxide fuel. This is because the uranium has been replaced with an inert diluent (zirconium oxide) resulting in a heavy-metal fuel loading about one-tenth that of the MOX fuel assembly; therefore, a much higher fuel consumption level (atom% burnup) is achievable in the NFF core for a similar fissile loading. Results show that the heavy-metal consumption level using NFF is about 52% in a single pass through the ALWR tier, much higher than the 5.17% with only 14% – 30% TRU consumption for the standard MOX fuel approaches.

Finally, detailed mass-flow data were generated for each approach with the ORIGEN2 code. An auxiliary program that combines the output data from the WIMS8 and ORIGEN2 runs for a given approach was created. This was necessary because we do not have the appropriate ORIGEN2 cross-section libraries corresponding to the advanced fuel forms being considered in this study. Because of the high burnup in the thermal spectrum of the ALWR, some heavy-metal cross sections vary significantly with burnup. The impacts of these variations could be adequately captured in ORIGEN2 calculations using the appropriate burnup-dependent cross-section data files. These variations are dependent on the core configuration, loading and operating conditions, and therefore a lot of effort would be required to generate the cross sections for the various approaches of interest in this study.

To preclude addressing this issue at the current time, an approach was devised to ensure that the most accurate TRU vector and heavy-metal consumption levels are transmitted to the other groups involved in this study. For a given approach, the WIMS8 and ORIGEN2 codes are executed using input data appropriate for the approach. The WIMS8 run tracks all the pertinent heavy-metal nuclides and fission products (about 100 of them) that account for 99% of the reactivity impact of the fission products. The ORIGEN2 run tracks a larger number of heavy-metal and fission-product isotopes, as well as additionally light and structural elements. In the ORIGEN2 run, one of the existing cross-section libraries (for a plutonium recycle core) is employed, however. The auxiliary code uses the ORIGEN2 output as base, stores the nuclide masses and other quantities, and then uses masses of the heavy-metal and fission nuclides present in the WIMS8 run to replace those in the ORIGEN2 output. The fission-product masses are constrained to ensure that the total fission-product mass is consistent with the total heavy metal consumed. Once these isotope masses have been replaced, the auxiliary program recalculates the radioactivity and thermal power to account for the impact of the new masses on these quantities.

Transmutation Advanced Gas Reactor (GT-MHR)

The reference results for the gas-cooled reactor Tier 1 options were based on detailed three-dimensional calculations with the MCNP Monte Carlo code, coupled with the MONTEBURNS burnup code (based on coupling MCNP with ORIGEN). The models were based on the configuration described in Appendix F, and represents a GT-MHR system operating at 600 MWt with 108 *fuel elements* in three annular rings with inner and outer rings of reflector blocks, as is shown in Figure F-2 of Appendix F. Each fuel element was modeled explicitly with 216 fuel channels and interspersed coolant passages, as is shown in Figure K-1; no burnable poison was assumed. Each fuel channel contained explicitly modeled coated-particle fuel in a graphite matrix at a 2.5% packing fraction, with particles assumed to be arranged in a body-centered cubic array. The particle dimensions and constituents are also given in Appendix F.

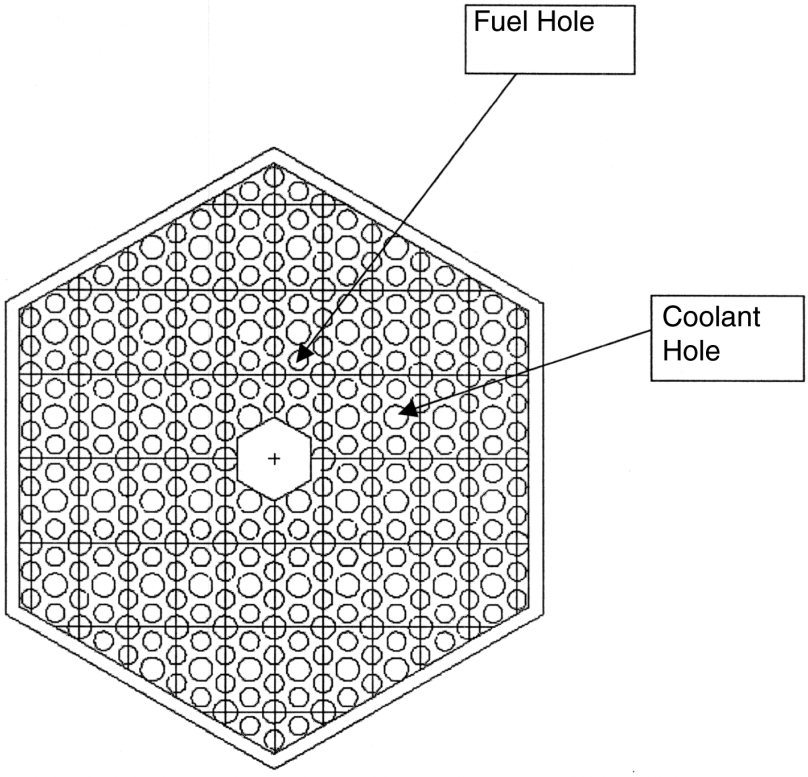


Figure K-1. MCNP/MONTEBURNS model for GT-MHR fuel block.

Two input material compositions were considered: all the transuranics (TRU) from commercial ALWR fuel (~50GWd per metric ton) that had been allowed to cool for 10 years, and only the plutonium isotopes from the spent fuel. The detailed distribution of isotopes for these compositions is given in Table K-1. The resultant total fuel loading in the core is 515 kg. The analyses assumed a single fuel batch because of time constraints. Analyses performed by General Atomics, as well as earlier studies by Argonne National Laboratory, show the potential improvements achievable with multi-batch fuel management strategies; it should be noted that these are fairly modest, confirming the adequacy of the analyses performed.

Burnup calculations were performed with MCNP/MONTEBURNS/ORIGEN for both fuel compositions until the value for k_{eff} dropped below 1.0; this defined the end-of -cycle burnup and isotopics. The calculation then continued in decay mode for 2, 5, 10, 15, and 20 years. Results were obtained on both an isotopic and element basis for activation products, actinides and daughters, and fission products for:

- Concentrations in grams;
- Activity in curies;
- Thermal power in watts;
- (Alpha,n) neutron source in neutrons/sec; and
- Spontaneous fission neutron source in neutrons/sec.

These results were provided to the separations team and as input to the Tier 2 evaluations.

The key results are summarized in Table K-3. Results from analyses performed by General Atomics using their standard design methodology (based on deterministic methods) are also presented for comparison; note, however, that they do not assume identical initial fuel compositions or core characteristics, as was assumed in the Monte Carlo analyses.

Table K-3. Comparison of GT-MHR System Performance Using MCNP/MONTEBURNS and Standard General Atomics Design Methodologies

	Pu-Separation: MA Bypass		No Pu Separation	
	MCNP/MONTEBURNS	**GA**	**MCNP/MONTEBURNS**	**GA**
Reactor System Thermal Power	600	2400 (4×600)	600	2400 (4×600)
Total Number of Fuel Assemblies	108×10	1020	108x10	340
Number of fuel batches	1	1	1	3
Cycle Length, efpd	500	840	400	240
Fuel Form	TRISO	TRISO	TRISO	TRISO
BEGINNING-OF-CYCLE Heavy Metal Inventory (t)	0.51	3.72	0.51	1.20
Charge per Batch				
Heavy Metal (HM) (t)	0.51	3.72	0.51	300
TRU in HM (%)	100	100	99.7	100
Pu in HM (%)	100	100	86.34	88.8
Fissile in HM (%)	64.37	71	55.64	61.2
Consumption per Batch				
Heavy Metal (HM) (%)	57.4	58.4	45.7	0.0
TRU (%)	57.4	58.4	46	54.8
Pu (%)	60.7	63.8	50.3	60.8
Fissile (%)	84.6	85.9	78.1	83.0
Discharge Burnup (MWd/kg)	591	542.6	470	480
Post-Irradiation Cooling period (yr)				
Average Linear Power (W/cm)	32.4		32.4	
Burnup reactivity loss (%delta-k)	33.9		25	

Accelerator-Driven Fast-Spectrum System (ATW)

Analyses of the fast-spectrum systems have focused on the equilibrium fuel cycle. The multiple-strata study assumes a steady-state feed of commercial spent fuel for the transmutation enterprise, and equilibrium system performance should be a good basis for comparing transmutation performance of different fuel-cycle strategies. Equilibrium-cycle performance characteristics were calculated using the REBUS-3 fuel-cycle analysis code. The fuel-cycle model accounts for repeated recycle in the fast-spectrum system and explicitly models the external cycle losses and hold-up times (e.g., accounts for radioactive decay during processing).

In the equilibrium fuel-cycle model, the charged fuel contains the transuranics recovered via recycling from the discharged ATW fuel, supplemented by either the Tier 1 discharge (in the double-tier approaches) or the commercial ALWR discharge TRU (Approach 3) to make up for the TRU consumed by fission. Determination of the equilibrium composition explicitly accounts for

0.1% loss of TRU in a two-year turnaround processing time. Also, 5% of the rare-earth fission products are carried over by the recycled fast-system TRU. The TRU mass loading in the fuel, which meets a targeted subcriticality level of 0.97 at beginning-of-equilibrium-cycle (BOEC), was determined using the REBUS-3 enrichment-search techniques. REBUS-3 also computes both batch-dependent and batch-averaged compositions at BOEC and end-of-equilibrium-cycle (EOEC) for each specified depletion region. In this study, five (equal length) axial depletion zones were consistently used; in the planar dimension, depletion zones consisted of individual fuel assemblies or groups of neighboring assemblies with similar reaction rates. Irradiation swelling of the metal fuel was modeled in the depletion calculations as a uniform 5% axial expansion of the fresh fuel, based on experiments for ternary metal (uranium-based) fuel.

Preliminary sensitivity studies of the effect of various flux computational options available in REBUS-3 were performed as part of previous ATW blanket design studies. Solutions obtained with nodal diffusion methods in hexagonal-Z geometry, the finite difference options in triangular-Z and R-Z geometries, and the VARIANT P1 approximation in hexagonal-Z geometry (ANL-95/40) were compared. Both the inhomogeneous source problem and the corresponding homogeneous eigenvalue problem (i.e., a system without the spallation source made artificially critical by use of an eigenvalue-to-scale neutron production) were considered.

The flux-solution sensitivity studies demonstrated that the global performance parameters are very similar for the different flux-calculation methods. The integral parameters and compositions estimated with the eigenvalue calculations were also found to agree well with the results of the corresponding inhomogeneous source calculations; peak flux and power were not as accurately predicted by the eigenvalue calculations. For computational convenience, therefore, homogeneous (eigenvalue) neutronic calculations performed using the hexagonal-Z nodal diffusion option of DIF3D were employed as a basis for assessing the transmutation performance of the ATW systems in this study. It is noted that the increase in source strength required to compensate the lower EOEC neutron multiplication can lead to large increases in flux in the vicinity of the source region; thus, inhomogeneous (source) calculations would be required to accurately predict the power peaking of the ATW systems.

Several refinements to the ATW computational methodology were employed in this study. The feed materials were prioritized so that the makeup feed (which has a higher fissile content than the recycle feed) preferentially loads into the outer fuel region (see Figure F-3 in Appendix F); this helps alleviate the central power peaking. The burnup chain was extended to model actinides ranging from uranium-234 to curium-248. A new metal fuel form was also considered. The TRU-40Zr alloy for this study has a much higher density than the dispersion fuel considered previously, so the assembly design had to be modified (lower fuel volume fraction) to achieve the desired fuel composition. New multi-group cross-section data were also generated for the nonfertile fast system. All previous results for the lead-bismuth eutectic (LBE) and sodium-cooled blanket studies have used a group-constant set developed for the ALMR pure burner design (*Nuclear Science and Engineering*, 1995), which employed a fixed hafnium poison. Since the detailed isotopic transmutation and losses are being tracked in the current study, it is essential to accurately model the self-shielding for a more representative composition and operating environment. For the ALWR feed directly into the nonfertile system (Approach 3), 21 group constants were generated with the MC^2-2 code for the inner core, outer core, reflector, target, and shield compositions. Based on preliminary results, these new group constants should be applicable for the entire range of nonfertile approaches. Finally, the commercial ALWR feed specified in Appendix F was employed. The performance results with these model refinements are compared to the previous sodium-cooled system point design results in Table K-4.

Table K-4. Comparison of Subcritical Fast-System Performance

Parameter		FY00 SPD	Approach 3M
BOEC heavy metal inventory (kg)		2620	2708
Fuel enrichment (weight % TRU in matrix)	Inner zone	31	59
	Outer zone	38	70
Multiplication factor	BOEC	0.970	0.969
	EOEC	0.920	0.928
Cycle length, days		135	140
Burnup reactivity loss (%Dk)		4.94	4.14
Peak linear power (W/cm)		397	385
Discharge burnup (MWd/kg)		275	273
Peak fast fluence (10^{23} n/cm^2)		4.06	3.73

The significant change in feed isotopics between the commercial ALWR discharge and mixed Tier 1 output requires an extra iteration in the equilibrium depletion computations. The reduced fissile content of the heavy metal requires a significant increase in the TRU inventory, compared with the approach where the commercial ALWR discharge feed is fed directly into the ATW system. The TRU content of the fuel cannot be changed, however, without adversely impacting the fuel properties, so the higher TRU content was achieved by increasing the fuel pin size (effectively increasing the fuel volume fraction) with a corresponding decrease in coolant volume fraction. In all approaches the resulting fuel volume fraction is still below the close-packed lattice used in conventional fast-reactor systems.

For the fast system analyses, detailed isotopic mass-flow data was generated using the ORIGEN-RA code; this local version of ORIGEN allows detailed specification of the initial isotopic composition and overwriting of the one-group cross-section data. Three separate ORIGEN computations were performed for the Approach 3 ATW system: the fuel region (including the alloy zirconium and bond sodium), in-core structural materials (clad, wire wrap, and duct in the active core zone), and ex-core structural materials (extended clad, wrap, and duct, pin end caps, lower shield, etc.) The two structural calculations are used to estimate the activation products. For these computations, the base ORIGEN-RA fast-reactor cross-section library is used with irradiation at the average core flux level (from Approach 3 with nonfertile fuel) for an average assembly lifetime (~7.5 cycles). The structural calculations were performed for the materials present in a single assembly. For this study, the assembly volume and structural volume fraction was conserved for all approaches, but the heavy metal inventory will vary with the incoming isotopic mix; thus, the in-core and ex-core structural mass-flow results will be used for all approaches. These results must then be normalized by the amount of transuranics (TRU) present in a single assembly.

The ORIGEN computation for 1 metric ton of fuel material is performed in more detail. The appropriate one-group cross sections and fluxes are obtained from the REBUS-3 depletion calculation. Scoping studies were performed using the BOEC and EOEC average cross-section values for the total core. It was demonstrated that the variation in discharge isotopics is minor; thus, the BOEC one-group cross sections will be utilized for convenience. Note that the approach-specific cross sections are required to reproduce the REBUS isotopic distributions. If the base ORIGEN cross-section data is used, the total burnup varies by only ~ 2% but the isotopic fractions vary by 10% to 50%. These isotopic variations will be very important for the

waste-characteristic evaluations targeted in this study. Scoping studies were also performed using the BOEC and EOEC total flux values; it was demonstrated that the average flux is adequate.

Transmutation Fast Reactor (ALMR)

For the transmutation fast reactor, the computational techniques are nearly identical to those described for the ATW fast-spectrum system. The only differences of significance are:

- Cross-section data appropriate for a fertile-fuel system was used. In particular, the region-dependent multigroup cross sections used in the original ALMR conventional and fertile-fuel burner design studies were employed.
- The close-packed fuel pin lattice employed in the ALMR burner design was retained. In the enrichment search, the pin size was not varied, but the TRU to uranium ratio was changed to assure criticality throughout the equilibrium cycle.

Appendix L: Evaluation Process Details

As described in Section 4, mass flow data was captured at each separations and irradiation/cooling step. These data were normalized and used to derive the values required to assess each approach relative to the criteria, as outlined in Table 2-1.

Model Development

Reactor models were developed for commercial ALWR spent nuclear fuel, which serves as the feed for all approaches, as well as for Tier 1 and Tier 2. For Tiers 1 and 2, neutronic models were developed independently by two research teams.

The commercial SNF feed vector is based on the isotopic charge and discharge of the 1989 ORNL reactor model for a PWR with 50,000 MWd per metric ton burnup [ORNL/TM-11018, December 1989].

Within Tier 1, a single-infinitely-reflected-assembly ALWR model was used for the approaches employing MOX and Zr Oxide fuels, with parameters varying by fuel content. For example, the actinide concentration of the fuel is adjusted to give the desired K_{eff}. A unit lattice (assembly) analysis approach was used. The systems are assumed to employ a full-core loading of a single fuel form (e.g., 100% MOX fuel or NFF). A separate reactor model was developed for the TRISO-fuelled GT-MHR system by each of two research teams. The models were based on the configuration described in Appendix F, and represents a GT-MHR system operating at 600 MWt with 108 *fuel elements* in three annular rings with inner and outer rings of reflector blocks, as is shown in Figure F-2 of Appendix F. Each fuel element was modeled explicitly with 216 fuel channels and interspersed coolant passages.

Tier 2 efforts required separate models for the accelerator-driven and transmutation fast reactor systems. The ADS model is based on a sodium-cooled blanket with an LBE target. The fast reactor model is based on a fertile fuel ALMR system, similar to *actinide burning* fuel cycle strategies postulated in previous US advanced reactor studies (i.e., IFR program).

For each irradiation step, isotopic vectors for concentration, radioactivity, and thermal power were captured at charge, discharge, and post-cool. A combination of REBUS, DIF3D, DRAGON, MCNP, ORIGEN, and MONTEBURNS was used. Additionally, isotopic vectors were calculated based on assumed separations and fabrication losses.

Calculations

To derive values for the criterion assessments, numerous external data sources were required. These sources are itemized in Table L-1, below. When available, a secondary data source was used as part of data quality assurance.

Table L-1. Source for Constants Used in Calculations

Content	Primary Source	Secondary Source
Half-lives	NNDC "nuclear wallet card"	ORIGEN decay library from LANL
Radiotoxicity	ICRP ingestion toxicity	
Atomic mass excess (MeV)	NNDC "nuclear wallet card"	
Atomic mass	Calculated: A+(mass excess/931.48)	NIST experimental masses
MeV/disintegration	ORIGEN decay library end6dec	
Gamma fraction of MeV/disintegration	ORIGEN decay library end6dec	
Gamma watts	Calculated: gamma fraction * total watts	ORIGEN output for 3M case from BNL

Summary statistics were calculated to address each of the quantitative criteria. In all cases, $T_{1/2}$ is calculated in seconds, post-cool concentration and atomic masses are in grams, and head loads are in watts.

Normalization

Uranium Ore: Assuming the enrichment of low-enriched uranium fuel is 4.2w%, it takes 7.5 tons of natural uranium (with 0.72w% enrichment) to make 1 ton of fresh fuel. To facilitate comparisons, calculations are thus normalized to that of 7.5 metric tons natural uranium ore.

HM vs. TRU: In order to make appropriate comparisons among approaches or to spent nuclear fuel without transmutation, additional normalization factors must be applied. The ORIGEN results are given in units of one metric ton heavy metal charge, which includes both uranium and transuranic elements. These ORIGEN2 burnup calculations must be converted to 1MT TRU (the fertile fuel cases include the diluent, uranium, as heavy metal, so the TRU fraction of fuel is substantially lower in the results).

To compare to spent nuclear fuel, normalization is based on the fraction of TRU in spent nuclear fuel, the fraction of TRU in heavy metal for fertile fuels, the fraction of TRU sent to waste after separations (the rest is recycled), and the fraction of TRU being sent to that step from previous ones (i.e., not burned in Stage 1 or Tier 1 and excluding minor actinides for Approach 1 Tier 1 cases). To compare cases to each other for one metric ton of TRU processed, normalization is simply based on the fertile fraction (the ratio of grams of heavy metal to grams of TRU). Fertile fraction of fuel varies by approach. Table L-2 shows the various normalization fractions that can be used as well as the total product of those normalizations. For the total amount of material going to waste for one approach, the normalization factor for spent nuclear fuel must be applied to the result (in units of toxicity, dose, or heat load), the normalization factor for Tier I waste must be applied to its result, and the normalization factor for Tier II waste must be applied to the results for Tier II waste. The sum of the three represents the final result for each approach.

Process Step: As each step assumes 1MT heavy metal as its input, the burnup and loss calculations are used to normalize the masses as they pass from one step to the next.

Table L-2. Normalization Factors for Discharge Results from ORIGEN Compared to High-Level Waste from ALWR Spent Nuclear Fuel

Tier 1	**1X**	**1X2**	**1Z**	**1G**	**2X**	**2Z**	**2G**		
Separations fraction to waste	0.001	0.001	0.001	0.001	0.001	0.001	0.001		
Fertile fraction (grams heavy metal to grams TRU)	11.32	7.35	1	1	5.49	1	1		
Fraction TRU from SNF sent to Tier 1	0.865	0.865	0.865	0.865	1	1	1		
Fraction TRU in SNF	0.0142	0.0142	0.0142	0.0142	0.0142	0.0142	0.0142		
Remaining material after Stage 1	1	0.7835	1	1	1	1	1		
Total Normalization	**0.000139**	**7.08E-05**	**1.23E-05**	**1.23E-05**	**7.8E-05**	**1.42E-05**	**1.42E-05**		
Tier 2	**1X**	**1XT**	**1Z**	**1G**	**2X**	**2Z**	**2G**	**3M**	**3T**
Separations fraction to waste	0.001	0.001	0.001	0.001	0.001	0.001	0.001	0.001	0.001
Fertile fraction (grams heavy metal to grams TRU)	1	2.61	1	1	1	1	1	1	3.09
Fraction TRU in SNF	0.0142	0.0142	0.0142	0.0142	0.0142	0.0142	0.0142	0.0142	0.0142
Fraction of remaining TRU after Tier 1	0.701716	0.701716	0.553314	0.50349	0.8613	0.482	0.54	1	1
# passes TRU goes to waste = 1/burnup	4.166667	5.678592	4.446421	4.882813	3.554924	4.314064	4.422822	3.418803	5.359057
Total Normalization	**4.15E-05**	**0.000148**	**3.49E-05**	**3.49E-05**	**4.35E-05**	**2.95E-05**	**3.39E-05**	**4.85E-05**	**0.000235**

Radiotoxicity

Radiotoxicity, as reported in this study, is calculated for each cooling-time-period of interest, for each approach, and is normalized to that of 7.5 metric tons natural uranium ore as described above. Total radiotoxicity is the sum of contributions from leakage from each separations and fuel fabrication step for initial ALWR spent fuel, Tier 1, and Tier 2. It is assumed that no residual material is left at the end of Tier 2 – all is burned; the waste comes from the separations processes between cycles.

Long-term ingestion toxicity was estimated for several main actinides (Np237, Pu238, Pu239, Pu240, Pu241, Pu242, Am241, Am242m, Am243, Cm242, Cm243, Cm244, Cm245, Cm246, Cm247, Cm248, Cf249). This proved to be an accurate estimate of total radiotoxicity from all isotopes out to 10,000 years when compared to an ORIGEN2 run. The concentrations and/or activities of the actinides were calculated and multiplied by radiotoxicity factors (in sieverts/ becquerel). These were then normalized relative to natural uranium ore. The equations used for each isotope were:

$$\text{Concentration}_{dY} = (\text{Post-Cool concentration}_d \times e^{-\lambda_d \times Y})$$

$$+ (\text{Post-Cool concentration}_p \times \lambda_p \times (e^{-\lambda_p \times Y} - e^{-\lambda_d \times Y})/(\lambda_d - \lambda_p))$$

where:

d = daughter = isotope of interest

p = parent from which buildup of daughter occurs

Y = the number of years post-cool

and λ = decay constant (1/time))

$$\text{Radiotoxicity} = \sum (\text{Concentration}_Y\ (g) \times \text{Radiotoxicity factor} \times SA) \div \text{Radiotoxicity}_{\text{NormalizedOre}}$$

where:

SA, in Bq/g= specific activity ~ $N_A \times \ln(2) \div (T_{1/2} \times \text{atomic mass})$

Radiotoxicity factor is in Sv/Bq

And $N_A = 6.023 \times 10^{23}$

The composition that was used for one metric ton natural uranium ore at equilibrium is given in Table L-3 below. These values were converted to becquerels (1 curie = 3.7×10^{10} becquerels) and multiplied by the radiotoxicity factor and summed over each isotope to get the total radiotoxicity. Total radiotoxicity was then normalized to natural uranium ore, as described above. Total radiotoxicity was then multiplied by 7.5 to get the total radiotoxicity of 7.5 metric tons of natural uranium ore.

Table L-3. Composition of One Metric Ton Natural Uranium Ore

Isotope	Activity (Ci)
Pb-210	3.40E-01
Ra-223	1.54E-02
Ra-226	3.40E-01
Ac-227	1.54E-02
Th-230	3.40E-01
Pa-231	1.54E-02
U-234	3.40E-01
U-235	1.54E-02
U-238	3.31E-01

The radiotoxicity factors used in this study were derived from ICRP equivalent committed dose factors for numerous different organs and radionuclides that estimate the number of cancer

deaths that might result from ingestion of a certain radionuclide.[4] These dose factors are then summed for each radionuclide using tissue-weighting factors that are based on data for radiation induced cancer deaths to give the effective dose. The resulting effective dose for each radionuclide is given in units of Sv/Bq.

The tissue weight factors include the probability of a fatal cancer per lethal dose augmented by the number of severe genetic disorders caused by unit dose, corrected for the number of non-fatal cancers, and multiplied by the ratio of the expected number of years of life lost for each fatal cancer to the average number of years of life lost for all fatal cancers. The tissue weighting factors used by the ICRP have an additional lethality fraction (probability for the total number of cancers) multiplied by the probability for non-fatal cancers, which is then added to the probability for fatal cancers. However, there does not appear to be justification for use of this lethality fraction, so the tissue weighting fractions used in this study were not multiplied by it. To verify that the results were correct, a weighted sum calculation was done using the ICRP-recommended tissue weighting fraction and the committed equivalent doses for each radionuclide. The results were compared to the effective doses calculated by the ICRP[5] and determined to be similar enough, giving justification to use the preferred tissue weighting fractions in the same calculation instead. These results were then used as radiotoxicity factors in this study.

Long-Term Dose

Dose is a sum of the contributions from all radionuclides, with those contributions varying independently over time. Examination of the Yucca Mountain Project TSPA-SR base case shows that only eleven radionuclides (U-234, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-243, Cm-244, and Cm-245) have individual dose peaks greater to or equal 1% of the total peak dose. These eleven radionuclides were grouped by decay-chain membership and the long-term contribution of each chain, relative to direct spent nuclear fuel disposal, was calculated for each approach. For each decay chain and approach, the reduction factor is thus:

$$(\text{Concentration}_{\text{SNF}} - \text{Concentration}_{\text{Approach}}) \div \text{Concentration}_{\text{SNF}}$$

Because the same radioisotopes are tracked in both the SNF and Approach cases, the reduction factor is time-independent. Concentrations used in the calculations were taken from the ORIGEN2 post-cool output. The resulting reduction factors are presented in Section 6-2.

Long-Term Heat Load

Long-term heat load was estimated by calculating 500-year and 1000-year concentrations of important actinides (Np237, Pu238, Pu239, Pu240, Pu241, Pu242, Am241, Am242m, Am243, Cm242, Cm243, Cm244, Cm245, Cm246, Cm247, Cm248, Cf249), multiplying those by the heat load (in W/g) for each isotope, and taking the sum of those products. The contribution of all fission products was also included but was negligible. To get the concentrations of most fission products and some of the actinides, straight decay could be used, but for others (most actinides plus Y-90 and Ba-137m, whose concentrations result from decay of Sr-90 and Cs-137

[4] "1990 Recommendations of the International Commission on Radiological Protection," ICRP Publication 60, *Annals of the ICRP*, v. 21, nos. 1–3, Pergamon Press, Oxford, 1991.

[5] "Age-dependent Doses to Members of the Public from Intake of Radionuclides: Part 5 Compilation of Ingestion and Inhalation Dose Coefficients," ICRP Publication 72, *Annals of the ICRP*, v. 26, no. 1, Pergamon Press, Oxford, 1996.

respectively), buildup from other isotopes was included. The following two equations were used, respectively:

Long-term heat load after Y years with straight decay for an isotope was calculated as:

$$\text{Post-Cool concentration} \times e^{-\lambda Y} \times \text{total watts/gram}$$

where

Y = number of years past post-cooling

the decay constant, λ, = $\ln(2) \times$ seconds per year $\div T_{1/2}$

Specific Activity (SA) in Bq/g = $\ln(2) * N_A \div (T_{1/2} \times \text{atomic mass})$

$J/MeV = 1.60219 \times 10^{-13}$

Watts/gram = MeV/disintegration (obtained from ORIGEN2) $\times$ SA $\times$ J/MeV

To include decay-chain buildup from other isotopes (Y-90, Ba-137m and most actinides), the equation is:

$$\text{Heat load}_d = \text{total watts/gram} \times (\text{Post-Cool concentration}_d \times e^{-\lambda_d \times Y})$$

$$+ (\text{Post-Cool concentration}_p \times \lambda_p \times (e^{-\lambda_p \times Y} - e^{-\lambda_d \times Y})/(\lambda_d - \lambda_p))$$

where d = daughter and p = parent.

Worker Dose

Worker dose through gamma and neutron sources is reported in gamma watts per metric ton TRU (normalized from one metric ton heavy metal as described previously), assuming a distance of one meter. Total worker dose is the sum of the gamma and neutron dose fraction of total heat load for all isotopes.

Post-Cool concentration $\times \gamma$ watts per gram, where, for each isotope:

γ Watts /gram = γ decay fraction $\times$ total watts/gram

The equation for calculating total watts per gram for each isotope is shown above, in the description of long-term heat load.

Waste Mass

Waste mass requiring deep geological disposal for each approach is compared directly to that of ALWR spent fuel. Waste estimates are calculated for transuranic content and fission-product content, in metric tons, and for waste volume, in cubic meters. The reduction factors are calculated as:

$$(\text{Concentration}_{SNF} - \text{Concentration}_{Approach}) \div \text{Concentration}_{SNF}$$

Plutonium Inventory

Plutonium inventory reduction is the ratio of the sum of SNF/Pu less Pu waste in Tier 1 less Pu waste in Tier 2, divided by SNF/Pu. This represents the percentage reduction of Pu sent to the repository. The reduction factors are calculated as:

$$(\text{PuConcentration}_{SNF} - \text{PuConcentration}_{Approach}) \div \text{PuConcentration}_{SNF}$$

Criticality risk is directly related to Pu inventory and thus assessments relative to this criterion are based on the plutonium inventory reduction factors calculated above.

Appendix M: Summary of Recent International Studies

Two recent sets of studies have strongly contributed to our understanding of potential transmutation studies in the United States: first, a set of French studies concerned with plutonium multi-recycle in pressurized water reactors (PWRs); and second, a very wide international study coordinated by the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) that attempted to compare the performances of several systems for transmuting transuranics.

CEA Study on Plutonium Multi-Recycling in PWRs

The official French waste management strategy has long been announced as a double-strata approach. The first, commercial stratum is concerned uniquely with generating electricity using either uranium- or plutonium-fueled reactors; enriched uranium would first be used in a PWR, and recycled plutonium would be used as part of mixed-oxide (MOX) fuel in a single pass in the same PWRs. Later recycles of plutonium would then be used in a commercial fast reactor. The second, noncommercial stratum would collect and transmute all minor actinides in an accelerator-driven system. This situation does not correspond to the French industrial situation, in which the development of fast reactors for commercial use has been significantly slowed down.

A first section of this strategy has already been implemented: the French are currently recycling plutonium once in 900-Mwe PWRs loaded with about 30% MOX assemblies. This approach slows the accumulation of plutonium, but it never reaches an equilibrium situation. In the absence of commercial fast reactors, the CEA (Commissariat à l'énergie atomique) has been studying various options for stabilizing their plutonium inventory through multi-recycling in PWRs. Three options have been analyzed and hold the promise of stabilizing this inventory:

- The MIX assembly [1] is a standard 17×17 assembly in which plutonium is blended with an enriched uranium support. The assembly is homogeneous. Studies have demonstrated that the MIX concept would be an effective plutonium management tool; nevertheless, after several recycles the uranium enrichment needs to be increased beyond the limit of current enrichment plants. Thus, the MIX concept is not regarded as a short-term solution.
- The APA concept [2] is an evolutionary 17×17 assembly: it contains standard enriched UO_2 fuel rods, together with large annular plutonium rods (PuO_2-CeO_2) providing a large local moderation ratio favorable to plutonium burning. Studies have demonstrated that this concept would be an excellent plutonium management tool leading to significant reduction in the asymptotic plutonium inventory; nevertheless, significant research and development is required before its implementation.
- The CORAIL concept [3] is also based on a standard 17×17 assembly, with a heterogeneous loading of MOX and UO_2 rods: 84 MOX rods are placed at the periphery of the assembly. Recent studies have shown that even after seven recycles, all major industrial criteria (e.g., reactor safety, peaking factors, uranium enrichment) are satisfied in a 1300-Mwe French PWR. The CORAIL concept does lead to a stabilization of the plutonium inventory, using only existing and well-demonstrated technologies.

It should be noted that all these cases lead to a stabilization of plutonium inventories (at different levels), but they can also contribute to increasing the minor actinide inventory when compared to a once-through or one-recycle scenario. Furthermore, recent preliminary information [4] also

indicates that recycling minor actinides in PWRs is not a viable solution, as it leads to significant difficulties in the fuel cycle.

The major conclusion of these studies is important for the future phases of the AAA system studies: it is possible, using mostly existing technologies, to stabilize the plutonium inventory. [Note that a direct extrapolation to the US situation must studied carefully, as it is not clear how many US reactors can practically be licensed for MOX assemblies.]

OECD Study

The Nuclear Energy Agency has recently launched a systems study [5] to compare fast-reactor (FR) and accelerator-driven systems (ADS) based transmutation systems with regard to reactor properties, fuel cycle requirements, economic aspects, and R&D needs. To evaluate the essential differences between these systems, an Expert Group composed of 36 experts from 14 countries and several international organizations selected a number of representative fuel-cycle strategies. Those strategies were then analyzed using well-validated computer codes and consistent data for reactor and fuel-cycle parameters.

The Expert Group selected eight schemes, including four transmutation schemes with fully closed fuel cycles that allow the essential differences between FR- and ADS-based transmutation strategies to be demonstrated. General assumptions were similar to those employed in the current US study; in particular, a recovery rate of 99.9% was assumed for all actinides. The eight case were:

- Light-water reactor (LWR) once-through (the current US disposal scheme);
- Plutonium burning in LWRs and multi-recycle in FRs, with geological disposal of minor actinides;
- Heterogeneous recycling, where plutonium and neptunium are multi-recycled in FRs, and curium and americium are recycled once as targets;
- Transuranic (TRU) burning in FRs, where the TRUs separated from spent fuel are sent to a closed FR fuel cycle;
- TRU burning in ADSs, where the TRUs separated from spent fuel are sent to a closed ADS fuel cycle;
- MOX-TRU burning, where a LWR MOX stage is used before introduction in the ADS system;
- Double strata, as described earlier; and
- FR strategy, where no LWRs are assumed, and only fast reactors are used for the multi-recycle of all TRUs.

The major conclusions of the study relevant to our current work are summarized in Reference 5 and are repeated here.

- All transmutation strategies with closed fuel cycles could, in principle, achieve high reductions in the actinide inventory and the long-term radiotoxicity of the waste. With respect to the reduction factors, the potentials of the FR and the ADS are very similar.
- The reductions are primarily determined by the fuel burnup and the reprocessing and fabrication losses.
- The *TRU Burning in FR* and the *Double Strata* schemes are similarly attractive. The former has the advantage that it can gradually evolve into a pure fast-reactor strategy, but it requires high initial investment in fast-reactor and advanced fuel-cycle technologies.

The latter confines minor actinide contamination to a very small part of the fuel cycle, which, however, calls for very innovative technology.

- Under comparable assumptions, the *Double Strata* and the *TRU burning in FR* strategies appear to be the most attractive transmutation strategies, both from technical and economical viewpoints.

References

1. Youinou, G., M. Delpech, A. Puill, S. Aniel, and J.L. Guillet, “Plutonium Management and Multirecycling in LWRs using a U-235 Support,” *Proceedings: GLOBAL 99*, Jackson Hole, Wyoming, 1999.
2. Puill, A., J. Bergeron, and M. Rohart, “Mastery of the Plutonium Inventory in PWRs: The APA Concept,” *Proceedings: GLOBAL 99*, Jackson Hole, Wyoming, 1999.
3. Youinou, G., A. Zaetta, A. Vasile, M. Delpech, M. Rohart, and J.L. Guillet, “Heterogeneous Assembly for Plutonium Multi-Recycling in PWRs: The CORAIL Concept,” *Proceedings: GLOBAL 2001*, Paris, France, 2001.
4. Salvatores, M., CEA, private communication.
5. Wydler, P. and L. Van Den Durpel, “Comparative Study of ADS and FR in Advanced Nuclear Fuel Cycles,” *Proceedings: GLOBAL 2001*, Paris, France, 2001.